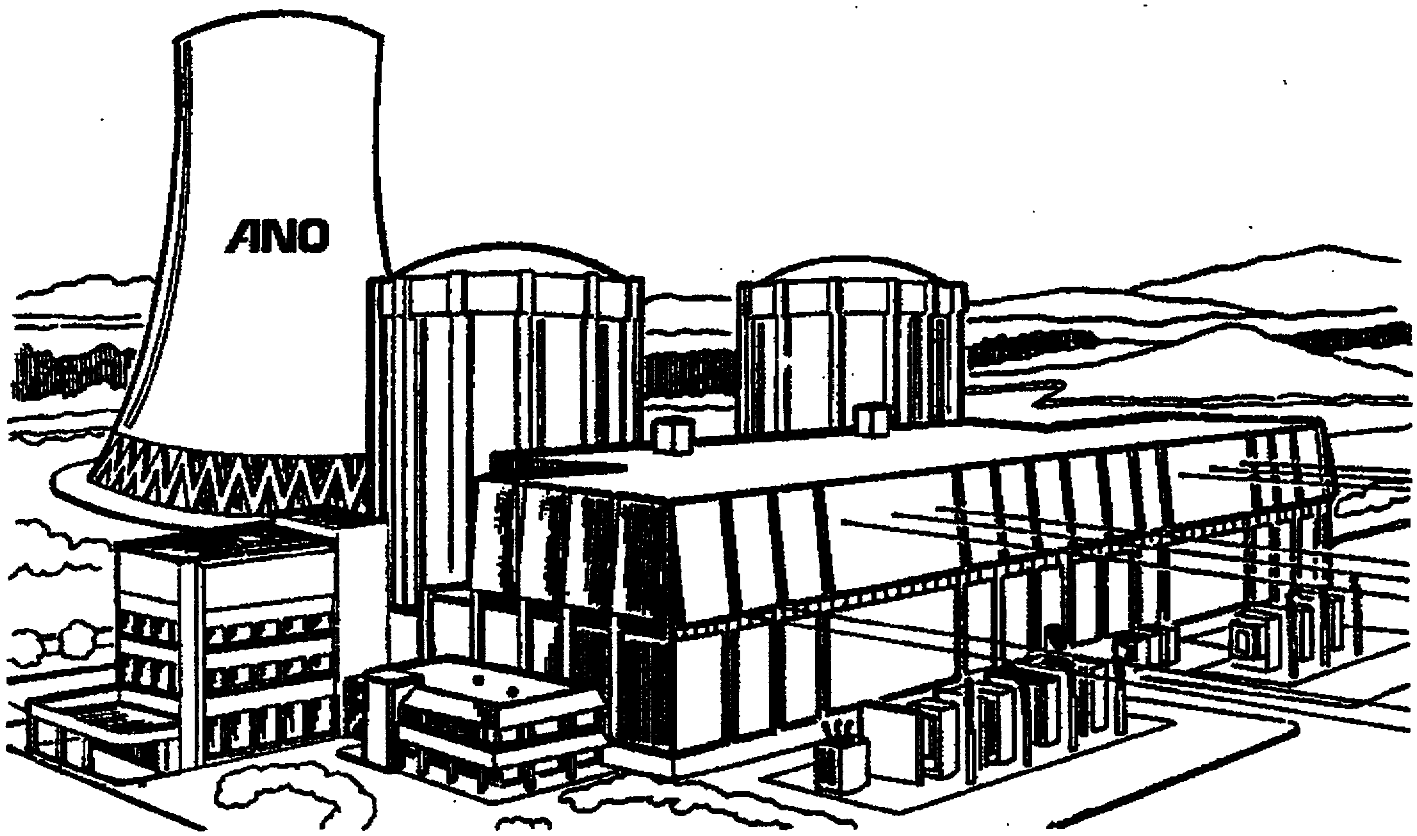
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL



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This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
3.6.1	3.6.1	Containment
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3.6 REACTOR BUILDING SYSTEMS

3.6.1 Reactor Building

LCO 3.6.1 The reactor building shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for reactor building air lock testing, in accordance with the Reactor Building Leakage Rate Testing Program.	In accordance with the Reactor Building Leakage Rate Testing Program

3.6 REACTOR BUILDING SYSTEMS

3.6.2 Reactor Building Air Locks

LCO 3.6.2 Two reactor building air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building," when air lock leakage results in exceeding the overall reactor building leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more reactor building air locks with one reactor building air lock door inoperable.	-----NOTE----- Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. ----------	
	A.1 Verify the OPERABLE door is closed in the affected air lock. AND	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed in the affected air lock. <u>AND</u> A.3 --------------NOTE-------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----------------------------------- Verify the OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
B. One or more reactor building air locks with reactor building air lock interlock mechanism inoperable.	------------------NOTES------------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of the reactor building is permissible under the control of a dedicated individual. --- B.1 Verify an OPERABLE door is closed in the affected air lock. <u>AND</u>	 1 hour

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	B.3	--------------NOTE-------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ------------------------------------ Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more reactor building air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall reactor building leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-----NOTE----- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Reactor Building Leakage Rate Testing Program.	In accordance with the Reactor Building Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.3 Reactor Building Isolation Valves

LCO 3.6.3 Each reactor building isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Penetration flow paths, except for purge valve penetration flow paths, may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for system(s) made inoperable by reactor building isolation valves.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ------------NOTE------------ Only applicable to penetration flow paths with two reactor building isolation valves. -------------------------------- One or more penetration flow paths with one reactor building isolation valve inoperable.	A.1 <u>AND</u>	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	48 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 -------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside the reactor building AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside the reactor building
B. -------------NOTE-------------- Only applicable to penetration flow paths with two reactor building isolation valves. ------------------------------------ One or more penetration flow paths with two reactor building isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------NOTE------------ Only applicable to penetration flow paths with only one reactor building isolation valve and a closed system. ----------------------------------- One or more penetration flow paths with one reactor building isolation valve inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	<u>AND</u>		
	C.2	------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each reactor building purge isolation valve is closed.	31 days

SURVEILLANCE		FREQUENCY
SR 3.6.3.2	------------------------------NOTE------------------------------ Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	------------------------------NOTE------------------------------ Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	Once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated reactor building isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.4 Reactor Building Pressure

LCO 3.6.4 Reactor building pressure shall be ≥ -1.0 psig and ≤ +3.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building pressure not within limits.	A.1 Restore reactor building pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify reactor building pressure is ≥ -1.0 psig and ≤ +3.0 psig.	24 hours

3.6 REACTOR BUILDING SYSTEMS

3.6.5 Reactor Building Spray and Cooling Systems

LCO 3.6.5 Two reactor building spray trains and two reactor building cooling trains shall be OPERABLE.

---NOTE---

Only one train of reactor building spray and one train of reactor building cooling are required to be OPERABLE during MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One reactor building spray train inoperable in MODE 1 or 2.	A.1 Restore reactor building spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One reactor building cooling train inoperable in MODE 1 or 2.	B.1 Restore reactor building cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
C. Two reactor building cooling trains inoperable in MODE 1 or 2.	C.1 Restore one reactor building cooling train to OPERABLE status.	72 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
E. One required reactor building spray train inoperable in MODE 3 or 4. <u>OR</u> One required reactor building cooling train inoperable in MODE 3 or 4.	E.1	Restore required inoperable train to OPERABLE status.	36 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1	Be in MODE 5.	36 hours
G. Two reactor building spray trains inoperable in MODE 1 or 2. <u>OR</u> Any combination of three or more trains inoperable in MODE 1 or 2. <u>OR</u> One required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4.	G.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify each reactor building spray manual, power operated, and automatic valve in each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.5.2	Operate each required reactor building cooling train fan unit for $\geq$ 15 minutes.	31 days
SR 3.6.5.3	Verify each required reactor building cooling train cooling water flow rate is $\geq$ 1200 gpm.	31 days
SR 3.6.5.4	Verify each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.5.5	Verify each automatic reactor building spray valve in each required flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.5.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.8	Verify each required train spray nozzle is unobstructed.	10 years

3.6 REACTOR BUILDING SYSTEMS

3.6.6 Spray Additive System

LCO 3.6.6 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each Spray Additive System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.2	Verify sodium hydroxide tank solution volume is ≥ 9000 gallons.	184 days
SR 3.6.6.3	Verify sodium hydroxide tank solution concentration is > 5.0 wt% and < 16.5 wt.% NaOH.	184 days

SURVEILLANCE		FREQUENCY
SR 3.6.6.4	Verify each Spray Additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.7 Hydrogen Recombiners

LCO 3.6.7 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombiner inoperable.	A.1 -------------NOTE-------------- LCO 3.0.4 is not applicable. ------------------------------------ Restore hydrogen recombiner to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Perform a system functional test for each hydrogen recombiner.	18 months
SR 3.6.7.2	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	18 months

SURVEILLANCE		FREQUENCY
SR 3.6.7.3	Perform a resistance to ground test for each heater phase.	18 months

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.1 Reactor Building

BASES

BACKGROUND

The reactor building consists of the reactor building (RB) structure, its steel liner, and the penetrations of this liner and structure. The reactor building is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, the reactor building provides shielding from the fission product radioactivity that may be present in the reactor building atmosphere following an accident.

The reactor building is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The reactor building design includes ungrouted tendons where the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The reinforced concrete structure is required for structural integrity of the reactor building under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the reactor building. Maintaining the reactor building OPERABLE limits the leakage of fission product radioactivity from the reactor building to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the reactor building boundary are a part of the reactor building leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either.

   1. capable of being closed by an OPERABLE automatic reactor building isolation system except as provided in LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," LCO 3.3.6, "ESAS Manual Initiation," and LCO 3.3.7, "ESAS Actuation Logic," or

   2. closed by manual valves, blind flanges, or de-activated automatic valves in their closed positions, except as provided in LCO 3.6.3, "Reactor Building Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Reactor Building Air Locks"; and

c. The equipment hatch is closed.

APPLICABLE SAFETY ANALYSES

The design basis for the reactor building is that the reactor building must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to the reactor building from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission products within the reactor building can occur from a LOCA. In the DBA analyses, it is assumed that the reactor building is OPERABLE such that, for a LOCA, fission product release to the environment is controlled by the rate of reactor building leakage. The reactor building was designed with an allowable leakage rate of 0.2% of reactor building air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as La: the maximum allowable leakage rate at the calculated maximum peak reactor building pressure (Pa) resulting from the limiting DBA. The allowable leakage rate represented by La forms the basis for the acceptance criteria imposed on all reactor building leakage rate testing. La is assumed to be 0.2% per day in the safety analysis at Pa = 54.0 psig (Refs. 2 and 3).

In MODES 1 and 2, the reactor building satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building satisfies Criterion 4 of 10 CFR 50.36.

LCO

Reactor building OPERABILITY is maintained by limiting leakage to within limits specified by the Reactor Building Leakage Rate Testing Program. Reactor building OPERABILITY for leakage is attained by ensuring that the equipment hatch and both doors of the personnel and emergency air locks are closed and sealed, except as appropriate for maintenance activities, and that the other isolation devices are closed, deactivated in the closed position, or OPERABLE as required. Reactor building OPERABILITY is also maintained by monitoring the deviation of key design parameters of the RB structure from the original design configuration and ensuring that structural limits are not exceeded. The structural monitoring programs are performed in accordance with Subsection IWL of ASME Section XI Boiler and Pressure Vessel Code, as referenced by 10CFR50.55a, and 10CFR50, Appendix J (Ref. 1). Compliance with this LCO, in conjunction with LCO 3.6.2, will ensure a reactor building configuration that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the reactor building air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the reactor building being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 La.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the reactor building is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from the reactor building. The requirements for the reactor building during MODE 6 are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

A.1

In the event the reactor building is inoperable, the reactor building must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining reactor building OPERABILITY during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring reactor building OPERABILITY) occurring during periods when reactor building is inoperable is minimal.

B.1 and B.2

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the reactor building OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Reactor Building Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless its contribution causes overall Type A, B, and C leakage to exceed limits. As left leakage prior to the first startup after performing a required leakage test is required to be ≤ 0.6 La for combined Type B and C leakage following an outage or shutdown that included Type B or C testing, and ≤ 0.75 La for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 La. At ≤ 1.0 La the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Reactor Building Leakage Rate Testing Program, which implements 10CFR50, App. J, and includes more frequent testing of the reactor building purge isolation valves. These periodic testing requirements verify that the reactor building leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J.
2. SAR, Chapter 14.
3. SAR, Section 5.2.
4. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.2 Reactor Building Air Locks

BASES

BACKGROUND

Reactor building air locks, also known as the personnel air lock and the emergency (or escape) air lock, form part of the reactor building pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder with a door at each end. The doors are interlocked to prevent simultaneous opening. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in the reactor building. As such, closure of a single door supports the reactor building OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in the reactor building internal pressure results in increased sealing force on each door).

The reactor building air locks form part of the reactor building pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the reactor building leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within the reactor building are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In the analysis of each of these accidents, it is assumed that the reactor building is OPERABLE such that release of fission products to the environment is controlled by the rate of the reactor building leakage. The reactor building was designed with an allowable leakage rate of ≤ 0.2% of the reactor building air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as La: the maximum allowable reactor building leakage rate at the calculated maximum peak reactor building pressure, (Pa), following a DBA (LOCA). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

In MODES 1 and 2, the reactor building air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building air locks satisfy Criterion 4 of 10 CFR 50.36.

LCO

Each reactor building air lock forms part of the reactor building pressure boundary. As a part of the reactor building, the air lock safety function is related to control of the reactor building leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test (i.e., closed and sealed), and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of the reactor building does not exist when the reactor building is required to be OPERABLE. Closure and sealing of a single door in each air lock provides sufficient leakage barrier following postulated events. Nevertheless, both doors are normally closed and sealed when the air lock is not being used for normal entry into or exit from the reactor building.

APPLICABILITY

In MODES 1, 2, 3, and 4, the reactor building air lock OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the reactor building air locks are not required in MODE 5 to prevent leakage of radioactive material from the reactor building. The requirements for the reactor building air locks during MODE 6 are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable but capable of being swung, then it and the air lock barrel may be easily accessed for most repairs. It is preferred that an inoperable inner door be accessed from inside the reactor building by entering through the other OPERABLE air lock. However, if this not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the reactor building boundary is not intact (during access through the OPERABLE door). Opening the OPERABLE door, even if it means the reactor building boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the reactor building during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall reactor building leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building."

A.1, A.2, and A.3

With one air lock door inoperable in one or more reactor building air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected reactor building air lock.

This ensures that a leak tight reactor building barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires the reactor building be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the remaining OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable reactor building leakage boundary is maintained. The Completion Time of once per 31 days is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by a Note that clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of this Note does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from the reactor building under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the reactor building inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), the reactor building remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a unit shutdown. In addition, even with both doors failing the seal test, the overall reactor building leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected reactor building air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that the reactor building be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining the reactor building air locks OPERABLE requires compliance with the leakage rate test requirements of the Reactor Building Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and reactor building OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall reactor building leakage rate. The Frequency is required by the Reactor Building Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of the Reactor Building Leakage Rate Testing Program. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C reactor building leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident reactor building pressure, closure of either door will support the reactor building OPERABILITY. Thus, the door interlock feature supports the reactor building OPERABILITY while the air lock is being used for personnel transit in and out of the reactor building. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the reactor building air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, and the potential for loss of reactor building OPERABILITY if the Surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not expected to be challenged during use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.

2. SAR, Chapter 14.

3. SAR, Chapter 5.

4. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.3 Reactor Building Isolation Valves

BASES

BACKGROUND

The reactor building isolation valves form part of the reactor building pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position, check valves, blind flanges, and closed systems are considered passive devices. Automatic valves designed to close following an accident without operator action, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically reactor building isolation valves) make up the Reactor Building Isolation System.

Reactor building isolation occurs upon receipt of a high reactor building pressure signal. The reactor building isolation signal closes automatic reactor building isolation valves in fluid penetrations not required for operation of engineered safeguard systems to prevent leakage of radioactive material. Also, upon receipt of a low RCS pressure signal, certain automatic reactor building isolation valves isolate. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the reactor building isolation valves (and blind flanges) help ensure that the reactor building atmosphere will be isolated in the event of a release of radioactive material to the reactor building atmosphere from the RCS following a Design Basis Accident (DBA).

OPERABILITY of the reactor building isolation valves (and blind flanges) supports the reactor building OPERABILITY during accident conditions.

The OPERABILITY requirements for the reactor building isolation valves help ensure that the reactor building is isolated. Therefore, the OPERABILITY requirements provide assurance that the reactor building function assumed in the safety analysis will be maintained.

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Reactor Building Purge System was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into the reactor building. The Reactor Building Purge System consists of one 24 inch line for exhaust and one 24 inch line for supply, with supply and exhaust fans. The reactor building purge supply and exhaust lines each contain two isolation valves that receive a reactor building isolation signal.

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large reactor building leakage path introduced by these 24 inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the reactor building design leakage rate of ≤ 0.2% of reactor building air weight per day (La) (Ref. 1). The 24 inch purge valves are not tested for automatic closure from their open position under DBA conditions. Therefore, the 24 inch supply and exhaust purge valves are maintained closed with the handswitch keys removed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the reactor building boundary is maintained.

APPLICABLE SAFETY ANALYSES

The reactor building isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the reactor building boundary during major accidents. As part of the reactor building boundary, the reactor building isolation valve OPERABILITY supports isolation of the reactor building. Therefore, the safety analysis of any event requiring isolation of the reactor building is applicable to this LCO.

The DBAs that result in a release of radioactive material within the reactor building are a loss of coolant accident (LOCA) and a main steam line break (Ref. 2). In the analysis for each of these accidents, it is assumed that the reactor building isolation valves are either closed or function to close. This ensures that potential paths to the environment through the reactor building isolation valves (including reactor building purge valves) are minimized. The safety analysis assumes that the 24 inch purge valves are closed at event initiation.

The LOCA analysis assumes a fixed amount of core inventory escapes. No mechanistic scenario is evaluated to determine what portion of the inventory is released prior to closure of the reactor building isolation valves. Industry standards for sizing valve operators govern the closure times of the reactor building isolation valves.

In MODES 1 and 2, the reactor building isolation valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, the reactor building isolation valves satisfy Criterion 4 of 10 CFR 50.36.

LCO

Reactor Building isolation valves form a part of the reactor building boundary. The reactor building isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the reactor building boundary.

The automatic isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 24 inch purge valves must be maintained closed. The valves covered by this LCO are listed in the SAR (Ref. 4).

Blind flanges are addressed within this Specification and treated as a separate type of manual "valve." The normally closed manual isolation valves are considered OPERABLE when valves are closed (or blind flanges are in place) or open under administrative control. These passive isolation valves/devices are listed in Reference 4.

The reactor building isolation valve leakage rates are addressed by LCO 3.6.1, "Reactor Building," as Type C testing.

This LCO provides assurance that the reactor building isolation valves will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the reactor building boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building isolation valves OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. The requirements for reactor building isolation valves during MODE 5 and 6, primarily related to movement of irradiated fuel in the reactor building, are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths with inoperable reactor building isolation valves, except for purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for reactor building isolation is indicated. Due to the size of the reactor building purge line penetration and the fact that those penetrations exhaust directly from the reactor building atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable reactor building isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable reactor building isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable reactor building isolation valve.

<u>A.1 and A.2</u>

In the event one reactor building isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic reactor building isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to the reactor building. Required Action A.1 must be completed within the 48 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting reactor building OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 48 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic verification is necessary to ensure that the reactor building penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside the reactor building and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside the reactor building" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside the reactor building, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two reactor building isolation valves. For penetration flow paths in closed systems with only one reactor building isolation valve, Condition C provides appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows the devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

With two reactor building isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of the reactor building and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two reactor building isolation valves. Condition A of this LCO addresses the condition of one reactor building isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one reactor building isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable, considering the relative structural integrity of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting reactor building OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure that reactor building penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one reactor building isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once verified to be in the proper position, is small.

D.1 and D.2

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 24 inch reactor building purge isolation valve in the purge system supply and exhaust is required to be verified closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of the reactor building is not caused by an inadvertent or spurious opening of a reactor building purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the closed position during MODES 1, 2, 3, and 4. A reactor building purge valve that is closed must have motive power to the valve operator removed. This can be accomplished by removing the valve handswitch key. The Frequency is consistent with other reactor building isolation valves discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each reactor building isolation manual valve and blind flange located outside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the reactor building boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those reactor building isolation valves outside the reactor building and capable of being mispositioned are in the correct position. Since verification of valve position for the reactor building isolation valves outside the reactor building is relatively easy, the 31 day Frequency was chosen to provide added assurance of the correct positions. The SR specifies that the reactor building isolation valves open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these reactor building isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.3

This SR requires verification that each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the reactor building boundary is within design limits. For reactor building isolation valves inside reactor building, the Frequency of "once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these reactor building isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that reactor building isolation valves open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these reactor building isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated reactor building isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period consistent with the industry standards for sizing valve operators. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.5

Automatic reactor building isolation valves close on a reactor building isolation signal to prevent leakage of radioactive material from the reactor building following a DBA. This SR ensures that each automatic reactor building isolation valve will actuate to its isolation position on a reactor building isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. SAR, Chapter 5.
2. SAR, Chapter 14.
3. 10 CFR 50.36.
4. SAR, Table 5-1.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.4 Reactor Building Pressure

BASES

BACKGROUND

The reactor building pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the reactor building design basis accident (DBA) and Emergency Core Cooling System (ECCS) analyses.

The reactor building pressure is a process variable that is monitored and controlled. The reactor building pressure limits are derived from the input conditions used in the reactor building DBA and ECCS analyses. Should operation occur outside these limits coincident with a DBA, post accident reactor building pressures and ECCS performance could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Reactor building internal pressure is an initial condition used in the DBA analyses to establish the maximum peak reactor building internal pressure. The limiting DBAs considered, relative to reactor building pressure, are the LOCA and SLB. The worst-case LOCA generates larger mass and energy release than the worst-case SLB. Thus, the LOCA event bounds the SLB event from the reactor building peak pressure standpoint (Ref. 1).

The initial pressure condition used in the reactor building analysis was 14.7 psia. The LCO limit of 3.0 psig ensures that, in the event of an accident, the design pressure of 59 psig for the reactor building is not exceeded. The LCO limit of -1.0 psig ensures that operation within the design assumptions for ECCS is maintained (Ref. 2). The LCO limit of 2.0 psig does not consider instrument uncertainty. The LCO limit of -1.0 psig is considered to be an as-indicated value.

For certain aspects of transient accident analyses, maximizing the calculated reactor building pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling Systems during the core reflood phase of a LOCA analysis increases with increasing the reactor building backpressure. Therefore, for the reflood phase, the reactor building backpressure is assumed in a manner designed to conservatively minimize, rather than maximize, the reactor building pressure response in accordance with 10 CFR 50, Appendix K (Ref. 3).

In MODES 1 and 2, the reactor building pressure satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building pressure satisfies Criterion 4 of 10 CFR 50.36.

LCO

Maintaining the reactor building pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak reactor building accident pressure will remain below the reactor building design pressure.

Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the ECCS analyses.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since maintaining reactor building pressure within design basis limits is essential to ensure that the peak reactor building pressure from an accident does not exceed the reactor building design pressure, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining the reactor building pressure within the limits of the LCO is not required in MODES 5 and 6.

ACTIONS

A.1

When the reactor building pressure is not within the limits of the LCO, the reactor building pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the reactor building analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Reactor Building," which requires that the reactor building be restored to OPERABLE status within 1 hour.

B.1 and B.2

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that the reactor building pressure is within limits ensures that operation remains within the limits assumed in the ECCS and the reactor building analyses. The 24 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of the reactor building pressure variations during the applicable MODES. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal reactor building pressure condition.

REFERENCES

1. SAR, Chapter 14.
2. SAR, Chapter 5.
3. 10 CFR 50, Appendix K.
4. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.5 Reactor Building Spray and Cooling Systems

BASES

BACKGROUND

The Reactor Building Spray and Reactor Building Cooling systems provide reactor building atmosphere cooling to limit post accident pressure and temperature in the reactor building to less than the design values. In the event of a Design Basis Accident (DBA), reduction of reactor building pressure reduces the release of fission products from the reactor building to the environment. The Reactor Building Spray and Reactor Building Cooling systems are designed to meet the requirements as discussed in the Safety Analysis Report (SAR), specifically, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal System," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Reactor Building Cooling System and Reactor Building Spray System are Engineered Safeguards (ES) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Reactor Building Spray System and Reactor Building Cooling System provide redundant reactor building heat removal operation. The Reactor Building Spray System and Reactor Building Cooling System provide redundant methods to limit and maintain post accident conditions to less than the reactor building design values.

Reactor Building Spray System

The Reactor Building Spray System consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes a reactor building spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ES bus. The borated water storage tank (BWST) supplies borated water to the Reactor Building Spray System during the injection phase of operation. In the recirculation mode of operation, Reactor Building Spray System pump suction is manually transferred to the reactor building sump.

The Reactor Building Spray System provides a spray of borated water into the upper regions of the reactor building to reduce the reactor building pressure and temperature during a DBA. During MODE 1 or 2, the Reactor Building Spray System supports the Spray Additive System function of iodine removal by providing the distribution mechanism. In MODES 3 and 4, sodium hydroxide is not mixed with the spray flow. In the recirculation mode of operation, heat is removed from the reactor building sump water by the decay heat removal coolers. Each train of the

Reactor Building Spray System provides adequate spray coverage to meet the system design requirements for reactor building heat removal.

The Reactor Building Spray System is actuated automatically by a reactor building High-High pressure signal. An automatic actuation opens the Reactor Building Spray System pump discharge valves and starts the Reactor Building Spray System pumps.

Reactor Building Cooling System

The Reactor Building Cooling System during normal operations consists of five (5) chilled water supplied cooling coils each in-line with a fan. Four (4) of these fan and chiller coil circuits have in-line service water cooling coils. During normal operations the service water to these coils is isolated. The post accident configuration of the Reactor Building Cooling System consists of the four service water cooling coils and their respective axial flow fans and dampers arranged as two independent trains.

Upon receipt of an Engineered Safeguards Actuation System (ESAS) RB high pressure signal, the four (4) fans associated with the service water coils receive a start signal, the chilled water is isolated, the service water supply and discharge valves open, the RB cooler bypass dampers open (which causes the return air to bypass the chilled water coils) and the RB cooler backdraft dampers receive an open signal. This equipment is powered from class 1E electrical power.

Each of the four (4) service water coil and fan air paths receives return air separately and directly from the RB atmosphere and discharges through ducting to a common plenum for distribution to the various reactor building spaces. The four (4) fans are mounted vertically on the ventilation units and are axial-flow type. The fan motors are single speed and operate in post-accident conditions at the same speed as normal conditions. Reducing fan motor speed during accident conditions is not required due to the reduced suction pressure drop (and hence fan load relative to normal conditions) created by bypassing the chilled water coils. An RB cooling train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation (Ref. 2). The continuous availability of appropriate service water flow to the RB Cooling System is assured by the periodic addition of a biocide to the Service Water System.

APPLICABLE SAFETY ANALYSES

The Reactor Building Spray System and the Reactor Building Cooling System reduce the temperature and pressure following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break. The postulated DBAs are analyzed, with regard to the reactor building ES systems, assuming the loss of one ES bus. This is the worst-case single active failure, resulting in one train of the Reactor Building Spray System and one train of the Reactor Building Cooling System being inoperable.

The analysis and evaluation show that, under the worst-case scenario, the highest peak reactor building pressure is 53.96 psig (experienced during a LOCA). The analysis shows that the peak reactor building temperature is 283.9°F (experienced during a LOCA). Both results are conservatively reported as 54 psig and 284°F, respectively, and are less than the design values. The analyses and evaluations assume a power level of 2568 MWt, one reactor building spray train and one reactor building cooling train operating, and initial (pre-accident) conditions of 140°F and 14.7 psia. The analyses also assume a delayed initiation to provide conservative peak calculated reactor building pressure and temperature responses.

The assumed Reactor Building Spray System total delay time of 300 seconds conservatively envelopes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, the reactor building spray pump startup, and spray line filling (Ref. 3).

The reactor building cooling train performance for post accident conditions is given in Reference 2. The result of the analysis is that each train can provide 100% of the required cooling capacity during the post accident condition. The train post accident cooling capacity under varying reactor building ambient conditions, is also shown in Reference 2.

The assumed Reactor Building Cooling System total delay time of 300 seconds conservatively envelopes signal delay, DG startup, block loading of equipment, fan startup, and service water pump startup times (Ref. 3).

In MODES 1 and 2, the Reactor Building Spray System and the Reactor Building Cooling System satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the Reactor Building Spray System and the Reactor Building Cooling System satisfy Criterion 4 of 10 CFR 50.36

LCO

During a DBA, the combination of one reactor building cooling train and one reactor building spray train is sufficient to reduce the reactor building pressure and temperature. One reactor building spray train is required, during MODE 1 or 2, to support the Spray Additive System in the removal of iodine from the reactor building atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, in MODES 1 and 2, two reactor building spray trains and two reactor building cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs. In MODE 3 or 4, one reactor building spray train and one reactor building cooling train are required to be operable. The LCO is provided with a Note which clarifies this requirement.

The Reactor Building Spray System includes spray pumps, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building sump.

The Reactor Building Cooling System includes cooling coils, dampers, axial flow fans, single speed fan motors, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since an event could cause a release of radioactive material in the reactor building as well as a temperature and pressure rise, the Reactor Building Spray System and the Reactor Building Cooling System are required to be OPERABLE in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Reactor Building Spray System and the Reactor Building Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one reactor building spray train inoperable in MODE 1 or 2, the inoperable reactor building spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to support the iodine removal and perform the reactor building cooling functions. The 72 hour Completion Time takes into account the redundant heat and iodine removal capability afforded by the OPERABLE reactor building train, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based on the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time.

B.1

With one of the reactor building cooling trains inoperable in MODE 1 or 2, the inoperable reactor building cooling train must be restored to OPERABLE status within 7 days. The remaining OPERABLE components are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time takes into account the redundant heat removal capabilities afforded by combinations of the Reactor Building Spray System and Reactor Building Cooling System and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action B.1 is based on the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time.

C.1

With two of the reactor building cooling trains inoperable in MODE 1 or 2, one of the reactor building cooling trains must be restored to OPERABLE status within 72 hours. The remaining spray system components (both spray trains are OPERABLE or else Condition G is entered) support iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time takes into account the redundant heat removal capabilities afforded by the Reactor Building Spray System and the low probability of a DBA occurring during this period.

D.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO, as modified by the Note, does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

E.1

With either one required reactor building (RB) spray train or one required reactor building cooling train inoperable in MODE 3 or 4, the inoperable train must be restored to OPERABLE status in 36 hours. The 36 hour Completion Time is reasonable based on consideration of the cooling capacity of the remaining required train of RB cooling or RB spray, the reduced reactor coolant energy in these MODES, and the short time spent in these MODES.

F.1

If the Required Action and associated Completion Time of Condition E are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 5 within 36 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

G.1

With two reactor building spray trains inoperable in MODE 1 or 2, or any combination of three or more reactor building spray and reactor building cooling trains inoperable in MODE 1 or 2, or one required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4, then LCO 3.0.3 must be entered immediately. The first part of this Condition addresses the loss of Spray Additive System support which would result from two inoperable reactor building spray trains in MODE 1 or 2. The second part of this Condition considers the loss of adequate reactor building cooling capacity in MODE 1 or 2 which would result from the loss of three or more of the four RB spray and RB cooling trains. Finally, the third part of this Condition addresses loss of reactor building cooling capability in MODES 3 and 4 when only one train of RB spray and one train of RB cooling are required.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the reactor building spray flow path provides assurance that the proper flow paths will exist for the Reactor Building Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside the reactor building and capable of potentially being mispositioned are in the correct position.

SR 3.6.5.2

Operating each required reactor building cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the redundancy available, and the low probability of a significant degradation of the reactor building cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.5.3

Verifying that a service water flow rate of 1200 gpm is provided to each required reactor building cooling train provides assurance that the original design flow rate is being achieved and that the service water flow rate is not degrading (Ref. 3). Assurance that the flow doesn't degrade by biological fouling between surveillances is provided by the addition of a biocide to the Service Water System whenever the service water temperature is between 60°F and 80°F. The Frequency was developed considering the known reliability of the system, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.5.4

Verifying that each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are measured during normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the Reactor Building Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test is indicative of overall pump performance. Such inservice tests confirm component OPERABILITY, trend performance, and may detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic reactor building spray valve actuates to its correct position and that each reactor building spray pump starts upon receipt of an actual or simulated actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.5.7

This SR requires verification by control board indication that each required reactor building cooling train actuates upon receipt of an actual or simulated actuation signal. The 18 month Frequency has been shown to be acceptable through operating experience. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.5.8

With the reactor building spray header isolated and drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this Surveillance demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the reactor building during an accident is not degraded. Due to the passive nature of the design of the nozzles, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. SAR, Section 1.4.
2. SAR, Chapter 6.
3. SAR, Chapter 14.
4. 10 CFR 50.36.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.6 Spray Additive System

BASES

BACKGROUND

The Spray Additive System reduces the iodine fission product inventory in the reactor building atmosphere resulting from a Design Basis Accident (DBA). The Reactor Building Spray System supports the Spray Additive System iodine removal function by providing a distribution mechanism for the solution.

The Reactor Building Spray System and Spray Additive System perform no function during normal operations. In the event of a loss of coolant accident (LOCA), the Spray Additive System will be automatically actuated upon a reactor building high-high pressure signal by the Engineered Safeguards Actuation System. Actuation of the Spray Additive System opens the sodium hydroxide isolation valves, which are powered from independent buses. When the valves are open, the sodium hydroxide solution is ready to be introduced into the RB Spray System headers.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of the dose consequences of an accident. It is absorbed by a sprayed solution from the reactor building atmosphere. The spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Sodium hydroxide (NaOH), because of its stability when exposed to radiation and elevated temperature, is the spray additive utilized.

The NaOH tank is designed and located to permit gravity draining into the Reactor Building Spray System. The sodium hydroxide volume requirement is given in gallons for compatability with the design analyses. The minimum NaOH tank volume of 9000 gallons preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level. Both the Reactor Building Spray System pumps initially take suction from the borated water storage tank (BWST) via two independent flow paths. The NaOH tank has a common outlet that splits and feeds each of the Reactor Building Spray System suction lines. The system is designed to discharge at a rate commensurate with the draining rate of the BWST so that all borated water injected is mixed with sodium hydroxide.

The flow rate is proportioned to provide a spray solution with a pH which is alkaline (Ref. 1). The range of alkalinity was established not only to aid in removal of airborne iodine, but also to minimize the corrosion of mechanical system components that would occur if the acidic borated water were not buffered. The pH range also considers the environmental qualification of equipment in the reactor building that may be subjected to the spray.

APPLICABLE SAFETY ANALYSES

The reactor building Spray Additive System provides for the effective removal of airborne iodine within the reactor building following a DBA.

Following the assumed release of radioactive materials into the reactor building, the reactor building is assumed to leak at its design value following the accident. The analysis assumes that most of the reactor building volume is covered by the spray.

The delay time assumed for the Spray Additive System is the same as for the Reactor Building Spray System and is discussed in the Bases for LCO 3.6.5, "Reactor Building Spray and Cooling Systems."

The LOCA analyses assume that one train of the Reactor Building Spray System/Spray Additive System is inoperable and that sufficient NaOH volume is added to the remaining BWST by the Reactor Building Spray System flow path.

In the evaluation of the worst-case LOCA, the safety analysis assumed that an alkaline reactor building spray effectively reduced the airborne iodine.

Each Reactor Building Spray System suction line is equipped with its own gravity feed from the NaOH tank. Therefore, in the event of a single failure within the Spray Additive System (i.e., NaOH isolation valve failure), NaOH will still be mixed with the borated water, establishing the alkalinity to provide effective iodine removal.

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the NaOH solution must be sufficient to provide NaOH into the spray flow until the Reactor Building Spray System suction path is switched from the BWST to the reactor building sump and to raise the long term sump solution pH to a level conducive to iodine removal. The long term sump solution pH is in the alkaline range. This pH maximizes the effectiveness of the iodine retention mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1 and 2, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Although the core is designed to retain structural integrity during an accident, fuel failure with resultant radioactive material release is postulated and the Spray Additive System is required OPERABLE in MODES 1 and 2.

In MODES 3 and 4, there is no postulated fuel failure contribution to radioactive material release and significantly less need for iodine removal capacity. Also, because of the limited time spent in these MODES, the probability of an event requiring use of the Spray Additive System is low. Therefore, the Spray Additive System is not required to be OPERABLE in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With the Reactor Building Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment capability of the spray solution for corrosion protection and iodine removal enhancement is reduced or non-existent in this Condition. The Reactor Building Spray System would still be available and would remove some iodine from the reactor building atmosphere in the event of a LOCA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst-case DBA occurring during this period.

B.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of NaOH manual, power operated, and automatic valves in the Spray Additive System flow path provides assurance that the system is able to provide NaOH to the Reactor Building Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside the reactor building capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

To provide the most effective iodine removal, the reactor building spray should be an alkaline solution. Since the BWST contents are normally acidic, the NaOH tank must provide a sufficient volume of NaOH to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The NaOH tank solution minimum volume of 9000 gallons corresponds to a tank level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. This parameter does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The minimum NaOH tank volume preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level. The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

SR 3.6.6.3

This SR provides verification of the NaOH concentration in the NaOH tank and is sufficient to ensure that the spray solution being injected into the reactor building is at the correct pH level. The concentration of NaOH in the NaOH tank must be determined by chemical analysis. There is no instrument uncertainty included in the surveillance limit values. Additional allowances for instrument uncertainty are contained in the implementing procedures. The 184 day Frequency is sufficient to ensure that the concentration of NaOH in the tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.6.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. SAR, Chapter 6.

2. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

Permanently installed hydrogen recombiners are required to reduce the hydrogen concentration in the reactor building following a loss of coolant accident (LOCA). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in the reactor building, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a LOCA.

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the control room, a power supply located in the auxiliary building, and a recombiner located in the reactor building. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at approximately 100 scfm. A single recombiner is capable of maintaining the hydrogen concentration in the reactor building below the 4 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safeguards (ES) bus and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in the reactor building to less than a concentration of 4 v/o following a DBA. This would prevent a hydrogen burn inside the reactor building, thus ensuring the reactor building pressure and temperature assumed in the accident analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate within the reactor building following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the reactor building sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to Reactor Building Spray System and Emergency Core Cooling Systems solutions.

To evaluate the potential for hydrogen accumulation in the reactor building following a LOCA, the hydrogen generation as a function of time following the initiation of the accident has been evaluated. Conservative assumptions presented by References 1 and 2 are used to maximize the amount of hydrogen calculated. These evaluations demonstrate approximately 8.9 days are needed for hydrogen concentration to increase to 4 v/o post LOCA without recombiner operation.

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst-case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, the hydrogen recombiner OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Two hydrogen recombiners are required OPERABLE in MODES 1 and 2 to assure control of hydrogen concentration within the reactor building to less than the flammability limit of 4 v/o.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an event requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in a reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

B.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can obtain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and power verified to be ≥ 60 kW. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.2

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.7.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. SAR, Section 6.6.
2. Regulatory Guide 1.7, Revision 2.
3. 10 CFR 50.36.

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification, NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or the NUREG. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 CTS 3.6.3 prohibits changing unit status where the reactor is made subcritical by less than 1% $\Delta k/k$ without reactor building (RB) integrity. This unit status is identified as ITS MODE 2 and is a Condition for which the RB is required OPERABLE. ITS LCO 3.0.4 adequately controls compliance with conditions required to be met for MODE changes. Therefore, CTS 3.6.3 is redundant and may be administratively deleted.

A4 Not used.

A5 CTS 3.6.6 is not specifically identified as applicable to reactor building (RB) penetrations with two (2) valves; however, testing and closure of the 'other' valve is discussed which implies that this was the intent. The NUREG 3.6.3 Condition A Note about applicability to systems with two RB isolation valves is explicit and is adopted as an administrative change in the ITS consistent with NUREG-1430.

A6 CTS 3.3.7(C) and (D) define Conditions where the requirements of CTS 3.3.4 (A) cannot be met because one or two trains of reactor building (RB) cooling are not OPERABLE while both trains of RB spray are OPERABLE. The NUREG 3.6.6 Conditions describe what is not OPERABLE but do not include what is OPERABLE since the LCO defines this. The requirements of ITS 3.6.5 Condition B are administratively equal to CTS 3.3.7(C). The requirements of ITS 3.6.5 Condition C are administratively equal to CTS 3.3.7(D). The CTS 3.3.7(C) and (D) statement that "but both reactor building spray systems are operable" is administratively deleted from the CTS to make the CTS Condition statement consistent with NUREG-1430.

A7 CTS 3.3.6 provides actions if the requirements of CTS 3.3.1 for one reactor building (RB) cooling train and one RB spray train are not met during MODES 3 and 4. CTS 3.3.7 (C), (D), and (E) are applicable for CTS 3.3.4 which is applicable during MODES 1 and 2. The references to CTS 3.3.7 and to reactor shutdown are administratively deleted from CTS 3.3.6 in accordance with the Applicability of ITS 3.6.5 Condition E.

A8 The NUREG 3.6.2 ACTIONS Notes 1, 2 and 3 are adopted by the ITS as appropriate statements of modifications to Conditions which are interpreted to be presently available in the CTS. Since these Notes are implied to be already available they are adopted as administrative changes.

A9 The NUREG 3.6.2 Condition A Note 1 is adopted administratively since the requirements of NUREG 3.6.2 Condition C are adopted for when two airlock doors are inoperable. This change is administrative because the present practice in implementing CTS 1.7a and b and CTS 3.6.1 for the personnel hatch and emergency hatch is equivalent.

A10 CTS 3.3.6 provides actions if the requirements of CTS 3.3.1 for one OPERABLE reactor building (RB) cooling train and one OPERABLE RB spray train are not met during MODES 3 and 4. However, CTS 3.3.6 does not provide an explicit time for restoration when the requirements of CTS 3.3.1 are not met. A restoration Completion Time of 36 hours is administratively adopted for ITS 3.6.5 Condition E and inserted appropriately in CTS 3.3.6. This is administrative since the sum of the Completion Times for ITS 3.6.5 Condition E and ITS 3.6.5 Condition F is equal to the time period provided by CTS 3.3.6 to be in MODE 5 if the requirements of CTS 3.3.1 are not met.

A11 Not used.

A12 CTS 4.5.2.1.2 (a)(1) Note 1 is administratively deleted since the "effective until" date of July 14, 1995, has been passed.

A13 CTS 4.12.1 b.2. indicates the visual examination of the hydrogen recombiners is looking for evidence of abnormal conditions "within" the recombiner enclosure. NUREG SR 3.6.8.2 (adopted as ITS SR 3.6.7.2) is worded slightly different in that it does not include the word "within" but still describes a visual examination which is considered administratively equivalent to the CTS examination. Since the two examinations are considered equivalent the ITS will adopt the NUREG SR 3.6.8.2 wording and remain consistent with NUREG-1430.

A14 CTS 3.3.7(E) describes a condition where one of two trains of reactor building (RB) spray and one of two trains of RB cooling are inoperable while meeting CTS 3.3.4(A), that is, during MODE 1 or 2. Since in the ITS, multiple Conditions of an LCO may be entered, the Condition of CTS 3.3.7(E) is equivalent to entering NUREG 3.6.6 Condition A (ITS 3.6.5 Condition A) concurrently with NUREG 3.6.6 Condition C (ITS 3.6.5 Conditon B). The equivalencies extend to the Required Actions and the Completion Times required in the CTS and the NUREG. The requirements of NUREG 3.6.6 Condition A and NUREG 3.6.6 Condition C are therefore adopted administratively as modified for ITS LCO 3.6.5 to retain consistency with NUREG-1430.

A15 Not used.

A16 CTS Table 4.1-2 Item #8 requires that the Reactor Building Isolation Trip be tested for "Functioning" every 18 months. The CTS functioning test is considered administratively equal to the NUREG SR 3.6.3.7 requirements to verify that each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal. NUREG SR 3.6.3.7 will be renumbered and adopted as ITS SR 3.6.3.5.

A17 CTS 4.4.1.4 does not specifically identify that isolation valves undergoing functional test are timed while being stroked. However, the CTS functional test is done in accordance with ASME Section XI which includes timing when testing valves. Therefore, NUREG SR 3.6.3.5, which is renumbered and adopted as ITS SR 3.6.3.4, is considered as administratively equivalent.

A18 CTS 1.7 conditions a., b., and c. provide configuration details concerning the OPERABILITY of the equipment hatch, the personnel and emergency hatches and non-automatic reactor building isolation devices. Since these details describe the OPERABLE configuration, they are administratively equivalent to stating that the equipment is OPERABLE. ITS LCOs 3.6.1, 3.6.2, and 3.6.3 will adopt the NUREG convention that the equipment be OPERABLE.

A19 The CTS 1.7 condition c. reference to "non-automatic reactor building isolation valves" is considered to envelope manual valves and check valves used as reactor building isolation devices. NUREG-1430 LCO 3.6.3 refers to these valve types individually. The CTS 1.7 condition c. definition is revised so that it is consistent with the NUREG and the administrative equivalence is evident.

A20 Fulfilling the requirements of the Reactor Building Leak Rate Testing Program (RBLRTP) is the equivalent of fulfilling the requirements of CTS 4.4.1 as required by CTS 1.7 condition e. The CTS 1.7 definition is revised to remain consistent with the NUREG for the ITS.

A21 This page is not yet approved as provided in this package. Therefore, this markup is dependent on the expected NRC approval of the August 6, 1998, (Ref. 1CAN089801) license amendment request (LAR) related to the sodium hydroxide tank level.

A22 Not used.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS 3.3.1 establishes the Applicability for a number of components, including the Reactor Building (RB) Spray System, the RB Cooling System, and the Engineered Safeguards System valves for these systems by referencing CTS 3.6.1. CTS 3.23.1 establishes Applicability for the RB purge valves similarly by referencing CTS 3.6.1. CTS 4.26.1 establishes Surveillance Frequencies for RB purge isolation valves relative to the requirement for RB Integrity (OPERABILITY) per CTS 3.6.1.

CTS 3.6.1 requires RB Integrity whenever all three following conditions exist:

a. Reactor coolant pressure is ≥ 300 psig,
b. Reactor coolant temperature is ≥ 200°F, and
c. Nuclear fuel is in the core.

With these criteria, RB Integrity would be required sometime during ITS MODE 4 but not necessarily when this MODE was entered from MODE 5.
The Applicabilities of NUREG 3.6.1 for RB OPERABILITY, NUREG 3.6.3 for the RB isolation valves and NUREG 3.6.6 for the RB Spray and Cooling Systems include MODES 1, 2, 3, and 4. This is an additional restriction on unit operation consistent with NUREG-1430.

M2 The CTS 3.6.1 progression of actions for failure to maintain reactor building (RB) integrity are to restore in 1 hour, be in hot standby (ITS MODE 2) in another 6 hours and in cold shutdown (ITS MODE 5) in a further 30 hours. NUREG 3.6.1 Required Action (RA) B.1, 3.6.2 RA D.1, and 3.6.3 RA D.1 require the unit to be in MODE 3 in 6 hours after entry into NUREG 3.6.1 Condition B. The NUREG requirement to be in MODE 3 (subcritical) rather than MODE 2 (critical) will be adopted in the ITS and is desirable in this instance because there is less potential energy in a non-critical reactor which could challenge RB OPERABILITY should an event occur. This is an additional restriction on unit operation consistent with NUREG-1430.

The CTS 3.6.4 progression of actions for failure to maintain RB pressure are identical to those for CTS 3.6.1 above. NUREG 3.6.4 RA B.1 requires the unit to be in MODE 3 in 6 hours after entry into NUREG 3.6.4 Condition B. The NUREG requirement to be in MODE 3 (subcritical) rather than MODE 2 (critical) will be adopted in the ITS for the reasons stated above. This is an additional restriction on unit operation consistent with NUREG-1430.

M3 CTS 3.6.4 and 3.6.6 set requirements while the reactor is critical. This unit status corresponds to ITS MODE 1 and 2. However, the CTS has an implied applicability of MODES 1, 2, 3 and 4 since these specs include a requirement to go to cold shutdown (ITS MODE 5) as part of the actions should the requirements not be met. During MODES 3 and 4, the reactor coolant is reduced to a temperature and pressure significantly below operating conditions at power. However, during these MODES, there remains sufficient stored energy within the coolant to allow any coolant released by a LOCA to flash to steam and thereby cause a release of fission products to the reactor building atmosphere. Although no core damage is anticipated due to a LOCA initiated during shutdown, the fission products present in the coolant at the time of the rupture would be available for release to the reactor building atmosphere. Therefore, maintaining reactor building OPERABILITY during MODES 3 and 4 ensures that the offsite radiation exposure of 10 CFR 100 is not exceeded. The Applicability of NUREG 3.6.4 and NUREG 3.6.6 is MODES 1, 2, 3, and 4. This Applicability will be adopted by the ITS to address the explicit and implicit requirements of the CTS. This is an additional restriction on unit operation consistent with NUREG-1430.

M4 CTS 3.6.5 requires a check of all (inside and outside) manual reactor building (RB) isolation valves "Prior to criticality following a refueling shutdown." The ITS will adopt NUREG SR 3.6.3.3 (renumbered to ITS SR 3.6.3.2) for position checks of valves outside the RB and NUREG SR 3.6.3.4 (renumbered to ITS SR 3.6.3.3) for position checks of valves inside the RB. The NUREG SR 3.6.3.3 requirements are to verify the position of appropriate valves outside the reactor building on a Frequency of 31 days. The NUREG SR 3.6.3.4 requirements place the inside valve position check Frequency as once when entering MODE 4 from MODE 5 unless done in the previous 92 days instead of when entering MODE 2 from MODE 3 as the CTS requires. The NUREG requirements are more consistent with the ITS threshold for RB OPERABILITY. These are additional restrictions on unit operation consistent with NUREG-1430.

M5 The NUREG 3.6.3 ACTIONS Notes 2 and 3 will be adopted in the ITS as appropriate clarifications of the ACTIONS for each reactor building (RB) isolation valve and its associated system. These details are not specifically addressed in the CTS.

The CTS has implied requirements, associated with requirements for RB integrity, which address two (2) inoperable valves in a penetration flow path, differentiate closed-system penetrations, or verify continued system isolation. The ITS will adopt NUREG 3.6.3 RA A.2 with Notes and both Completion Times, NUREG 3.6.3 Condition B with Note and NUREG 3.6.3 ACTION C with both Notes as appropriately specific, and therefore more restrictive, means of addressing requirements for RB isolation valves. These requirements are additional restrictions on unit operation consistent with NUREG-1430.

M6 Not used.

M7 CTS 3.3.4(B) provides the limits for volume and concentration for the sodium hydroxide (NaOH) tank. CTS Table 4.1-3 item #6 provides the surveillance Frequency for the NaOH solution concentration, however, there is no comparable NaOH tank solution volume surveillance Frequency requirement in the CTS. The ITS will adopt the NUREG SR 3.6.7.2 surveillance Frequency of 184 days for the NaOH tank solution volume to remain consistent with the surveillance Frequency of the NaOH tank solution concentration as well as the requirements of NUREG-1430.

M8 For CTS 3.3.6 and 3.3.7, in the event of concurrent reactor building (RB) spray or RB cooling train inoperability, the existing requirements allow independent application of allowed repair times without restriction. When a subsequent inoperability occurs just prior to restoration of the previous inoperability and close to the expiration of the CTS-allowed 36 hours for RB spray or 7 days for RB cooling, when taken to extreme, this independent application can provide an unlimited time of operation with an inoperable RB Spray or RB cooling train. While these simultaneous inoperabilities are expected to be rare, adoption of the maximum restoration time limit provided by NUREG 3.6.6 A.1 and C.1 is proposed to prevent extended operation in the respective Conditions. The proposed Technical Specifications format presents this as an additional Completion Time of "10 days from discovery of failure to meet the LCO" for both ITS 3.6.5 Required Action (RA) A.1 and RA B.1 and is considered to be reasonable. These additional Completion Time requirements represent additional restrictions on unit operation consistent with NUREG-1430.

M9 Not used.

M10 NUREG SR 3.6.7.1 requires surveillance of the position of the manual, power operated and automatic valves in the Spray Additive System on a frequency of 31 days. NUREG SR 3.6.7.4 requires surveillance of the actuation of the sodium hydroxide flow path automatic valves every 18 months. Neither of these surveillances is in the CTS, however, they are proposed to be adopted in the ITS as adequate methods, compatible with the system design, for assuring the availability of the Spray Additive System for its safety function. These NUREG requirements, adopted by the ITS, are additional restrictions on the operation of the unit consistent with NUREG-1430.

M11 In CTS 1.7 a & b, the air lock doors are required to be "closed and sealed" to establish reactor building integrity. In the CTS context, "sealed" means meeting leakage program requirements. The NUREG 3.6.2 Required Action (RA) A.2 requirements, however, include locking the door and the NUREG 3.6.2 RA A.3 requirements include verifying the door is locked. The NUREG requirements to lock the doors when performing these Required Actions will be adopted by the ITS.

The Note on NUREG 3.6.2 RA A.3 which provides that administrative means may be used to verify that air lock doors in high radiation areas are locked closed is adopted by the ITS.

The NUREG SR 3.6.2.1 Notes 1 and 2 are adopted as appropriate modifiers of the Surveillance Requirement. Furthermore, the results of air lock leakage testing, when performed, should be compared with the overall reactor building leakage from other sources to evaluate compliance with the Reactor Building Leakage Rate Testing Program (RBLRTP).

These NUREG requirements, adopted by the ITS, are additional restrictions on the operation of the unit consistent with NUREG-1430.

M12 The CTS requirements for the reactor building (RB) air locks (alternate ANO-1 terminology is the "personnel hatch" and the "emergency hatch") are provided by the RB integrity definition: CTS 1.7 a & b and by CTS 3.6.1 requirements for OPERABILITY.

The air lock door interlock requirements of NUREG 3.6.2 Condition B, including the Condition Notes 1 and 2, and the NUREG 3.6.2 Required Action (RA) B.3 Note are adopted by the ITS to provide specific guidance for this air lock feature. The CTS doesn't provide specific guidance for verification of air lock interlock function. NUREG Surveillance Requirement (SR) 3.6.2.2 will be adopted as appropriate and consistent with the significance of maintaining RB OPERABILITY.

The "reasons other than Condition A or B" air lock inoperable requirements of NUREG 3.6.2 Condition C are adopted by the ITS to provide specific guidance if the reason for an inoperable air lock is related to other than a door or interlock.

These NUREG requirements, adopted by the ITS, are additional restrictions on the operation of the unit consistent with NUREG-1430.

M13 CTS 3.6.4 addresses the reactor building (RB) internal pressure requirements but doesn't provide for surveillance of this parameter. It is proposed that NUREG Surveillance Requirement (SR) 3.6.4.1 be adopted in the ITS to require verification of RB internal pressure on a Frequency of 24 hours. Adoption of this Surveillance Requirement will replace present administrative verification of this parameter and provide appropriate ITS verification of safety analysis assumptions. This requirement is an additional restriction on the operation of the unit consistent with NUREG-1430.

M14 CTS 3.6.4 describes the reactor building (RB) internal pressure as not to exceed 3.0 psig or 5.5 inches Hg vacuum. Verifying RB pressure in inches of mercury, vacuum, is inconsistent with the NUREG format for reactor building (RB) pressure. The equivalent of 5.5 in. Hg vacuum is -2.7 psig, however, this value conflicts with the value assumed in the ANO-1 Emergency Core Cooling System (ECCS) analysis. To resolve these difficulties, the ITS will adopt the NUREG format for the RB pressure limits and the values consistent with the ECCS analysis. ITS LCO 3.6.4 and ITS SR 3.6.4.1 will provide RB pressure limits of $\geq$ -1.0 psig and $\leq$ +3.0 psig which is a lesser range than the CTS and is therefore more restrictive on unit operation. This more restrictive change makes the two limits of the range compatible with each other, with the control room indication of the RB pressure, with the appropriate analyses and with NUREG-1430.

M15 CTS 4.5.2.1.1 and CTS 4.5.2.2.2 describe requirements for Reactor Building (RB) Spray System and valve testing, however, there is no requirement for periodic verification of RB Spray System valve lineup. The ITS will adopt the requirements of NUREG SR 3.6.6.1 (as ITS SR 3.6.5.1) as an adequate method of verifying that the RB Spray System will be available if required. The Frequency of 31 days is consistent with the test frequency, in the CTS, of other portions of the RB Cooling System and Spray System. This surveillance is an additional restriction on the operation of the unit consistent with NUREG-1430.

M16 CTS 3.6.5 provides that the provisions of CTS 3.0.3 are not applicable. NUREG LCO 3.6.3 doesn't include such a note, so the requirements of NUREG LCO 3.0.3 are applicable. The requirements of NUREG LCO 3.6.3 will be adopted by the ITS since this LCO provides ACTIONS which will bring the unit to MODE 5 and place the unit in a condition which is more consistent with the ITS threshold for setting RB OPERABILITY. The requirement that LCO 3.0.3 is applicable for ITS 3.6.3 is an additional restriction on the operation of the unit consistent with NUREG-1430.

M17 Not used.

M18 CTS 3.3.6 provides that if the requirements of CTS 3.3.4 for two trains of reactor building (RB) cooling and two trains of RB spray and an OPERABLE Spray Additive System in MODES 1 and 2 can not be met then the unit shall be in MODE 3 within 36 hours. The ITS 3.6.5 Condition D and ITS 3.6.6 Condition B will adopt a Completion Time of 6 hours to be in MODE 3. This is an additional restriction on the operation of the unit consistent with NUREG-1430.

M19 CTS 3.3.6 provides actions if the requirements of CTS 3.3.1 for one OPERABLE reactor building (RB) cooling train and one OPERABLE RB spray train are not met during MODES 3 and 4. ITS 3.6.5 Condition E is comparable. CTS 3.3.6 requires that the unit be in MODE 5 in 72 hours if the conditions of CTS 3.3.1 can not be met. ITS 3.6.5 Condition F is comparable and will adopt a Completion Time of 36 hours to be in MODE 5. The response time reduction from 72 hours to 36 hours is an additional restriction on the operation of the unit which is consistent with the requirements NUREG-1430.

M20 CTS 3.3.1, CTS 3.3.4, CTS 3.3.5, CTS 3.3.6 and CTS 3.3.7 together form a matrix of requirements for the reactor building (RB) spray and RB cooling trains during MODES 1, 2, 3, and 4. This CTS matrix specifically describes some combinations of OPERABLE and inoperable trains of the two (2) trains of RB cooling and two (2) trains of RB spray during MODES 1 and 2 and the respective required trains during MODES 3 and 4. The CTS matrix, however, doesn't specifically address the combinations described in NUREG 3.6.6 Condition F or its modified version ITS 3.6.5 Condition G. Therefore, CTS 3.3.6 is considered to provide guidance for actions when conditions are not specifically described unless CTS 3.0.3 is considered appropriate. When implementing CTS 3.3.6, if the combination of inoperable trains described in ITS 3.6.5 Condition G were discovered, then CTS 3.0.3 would be considered appropriate and would be entered immediately which is the same Required Action provided by ITS 3.6.5 Condition G. However, the CTS 3.0.3 requirement to be in MODE 3 is 13 hours whereas the ITS LCO 3.0.3 requirement is 6 hours. This reduced response time is an additional restriction on the operation of the unit consistent with NUREG-1430.

M21 Not used.

M22 CTS 3.6.6 provides 24 hours to be in cold shutdown (ITS MODE 5) upon failure to restore an inoperable reactor building (RB) isolation valve or close the other valve after 48 hours while the reactor is critical (i.e., MODE 1 or 2). However, CTS 3.6.6 does not require a time period to be in an intermediate MODE on the descent to MODE 5 although the relationship of CTS 3.6.6 to CTS 3.6.1 by the reactor building integrity requirement implies MODE 3 in 6 hours. NUREG 3.6.3 Required Action (RA) E.1 requires that the unit be in MODE 3 in 6 hours on the way to MODE 5 from a similar Condition. Along with other NUREG requirements, the NUREG 3.6.3 RA E.1 requirements will be adopted as ITS 3.6.3 RA D.1, which is a more restrictive condition on unit operation consistent with NUREG-1430.

TECHNICAL CHANGE—LESS RESTRICTIVE

L1 Not used.

L2 Not used.

L3 The general CTS 3.3.5 maintenance requirements which are applicable to an inoperable reactor building (RB) spray system and the RB cooling system, are revised to be consistent with specific NUREG-1430 requirements for an inoperable RB spray train or RB cooling train. CTS 3.3.5 allows a train of these systems to be made inoperable for up to 24 hours for maintenance, but only if the redundant train is demonstrated operable within 24 hours prior to beginning the maintenance. However, the performance of maintenance on one train does not change the basis for believing that the redundant train is OPERABLE, therefore, this requirement is omitted from the ITS. This change is consistent with NUREG-1430.

L4 Not used.

L5 The requirements of CTS 3.3.7(C) for one inoperable reactor building (RB) cooling train, CTS 3.3.7(D) for two inoperable RB cooling trains, and CTS 3.3.7(E) for one inoperable RB spray and one inoperable RB cooling train correspond, respectively, to the Conditions of ITS 3.6.5 Condition B, Condition C, and a combination of Condition A and B. The Required Actions of these conditions, in the CTS, include time periods to be in hot shutdown (ITS MODE 3) and then cold shutdown (ITS MODE 5). The ITS will divide the Required Action to be in MODE 3 from the requirement to be in MODE 5 as appropriate for the Applicability of the Condition. These ITS Required Action requirements are less restrictive because the CTS required the unit to be placed in MODE 5 even though the Applicability of CTS 3.3.7(C), (D), and (E) is MODES 1 and 2.

L6 The requirements of CTS 3.3.6 (by reference to CTS 3.3.4) for one inoperable reactor building (RB) spray train or an inoperable Spray Additive System correspond, respectively, to the Conditions of ITS 3.6.5 Condition A and ITS 3.6.6 Condition A. The Required Actions of CTS 3.3.6 include time periods to be in hot shutdown (ITS MODE 3) and then cold shutdown (ITS MODE 5). The ITS will divide the Required Action to be in MODE 3 from the requirement to be in MODE 5 as appropriate for the Applicability of the Condition. These ITS Required Action requirements are less restrictive because the CTS required the unit to be placed in MODE 5 even though the Applicability of CTS 3.3.6, in this context, is MODES 1 and 2.

L7 CTS 3.14.1 requires that two hydrogen recombiners be OPERABLE whenever reactor building (RB) integrity is required, that is, during ITS MODES 1, 2, 3 and part of 4. The NUREG 3.6.8 Applicability is MODES 1 and 2. The NUREG Applicability considers that in MODES 3 and 4 both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, a requirement for OPERABLE hydrogen recombiners during MODES 3 and 4 will not be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L8 CTS 4.12.1a. requires verifying the OPERABILITY of each hydrogen recombiner system by system functional test at least once per 6 months. NUREG SR 3.6.8.1 (ITS SR 3.6.7.1) extends this Frequency to 18 months. Experience has shown that these components usually pass the Surveillance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This Frequency extension is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L9 CTS Table 4.1-3 item #6 requires that the sodium hydroxide tank solution concentration be sampled "quarterly and after each makeup." NUREG SR 3.6.7.3 requires a sampling Frequency of 184 days. The 184 day Frequency provided by the NUREG is sufficient to ensure that the NaOH concentration is within the established limits. This conclusion is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated from makeup sources) and the probability that any substantial variance in tank volume will be detected. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L10 CTS 4.5.2.1.1(b) requires that the availability of the reactor building (RB) spray headers and nozzles be verified at least every five (5) years. NUREG SR 3.6.6.8 requires that each spray nozzle be verified unobstructed on a Frequency of every ten (10) years. Due to the passive nature of the design of the nozzles, a ten (10) year Frequency is considered adequate to detect obstruction of the nozzles and will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L11 CTS 4.5.2.1.2(a) requires that the service water flow rate of each reactor building (RB) cooling train be tested on a frequency of at least once per 14 days. NUREG SR 3.6.6.3 requires a Frequency of 31 days. The Frequency of 31 days for testing of the service water flow is consistent with the CTS test frequency of other portions of the RB Cooling System. Furthermore, the service water supply to the RB cooling trains is considered reliable and there is a low probability of a significant degradation of flow occurring on a frequency of 31 days. Extension of this Frequency from 14 days to 31 days is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L12 The CTS requirements for RB penetrations do not provide alternate position verification methods for valves in high radiation areas. The ITS will adopt the NUREG SR 3.6.3.3 Note and the SR 3.6.3.4 Note allowing verification by administrative means for valves and blind flanges in high radiation areas. These allowances are adopted consistent with Industry ALARA practice while adequately addressing RB OPERABILITY requirements. Their adoption is a less restrictive condition on unit operation consistent with NUREG-1430.

L13 CTS 3.14 requires two (2) hydrogen recombiners to be OPERABLE whenever reactor building (RB) integrity is required, that is, during MODES 1, 2, 3, and part of 4. This direction implies that MODE changes are not allowed with only one (1) hydrogen recombiner OPERABLE. The ITS will adopt the note on NUREG 3.6.8 Required Action A.1 which makes NUREG LCO 3.0.4 not applicable when one hydrogen recombiner is inoperable. Accepting this Note will allow for MODE changes and unit operation during the Completion Time of 30 days adopted for ITS 3.6.7 Required Action A.1. This allowance is predicated on the availability of the other, 100% capacity, hydrogen recombiner, the small probability of a LOCA occurring and the amount of time available after a LOCA for operator action to prevent hydrogen accumulation from exceeding the flammability limit. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L14 CTS 3.3.4(C) requires that the manual valves in the sodium hydroxide tank main discharge lines shall be locked open before the reactor is made critical (i.e., ITS MODE 2). The NUREG LCO 3.6.7 requirement is an OPERABLE system, which implies that the manual valves be properly positioned but not necessarily locked. NUREG SR 3.6.7.1 requires that the manual valves (and the other valves) in the system, which are not locked or otherwise secured, be verified in their proper position. Therefore, the ITS will delete the requirement to have the Spray Additive System manual valves locked open. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L15 CTS 3.3.6, unless modified by CTS 3.3.7, provides that if the requirements of CTS 3.3.4 for two trains of reactor building (RB) cooling and two trains of RB spray and an OPERABLE Spray Additive System in MODES 1 and 2 can not be met then the unit shall be in MODE 3 within 36 hours. This 36 hour time period, without CTS 3.3.7 modification, is applicable and includes restoration time when one RB spray train or the Spray Additive System is inoperable in MODES 1 or 2. The ITS will adopt the NUREG 3.6.6 Condition A and the NUREG 3.6.7 Condition A Completion Time of 72 hours for restoration when these conditions exist (see DOC M8 for the requirement to be in MODE 3). This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L16 CTS 3.6.6 addresses the OPERABILITY of reactor building (RB) penetrations with two valves and includes the testing of the "other" valve when one of the valves is recognized as inoperable in a position other than closed. NUREG 3.6.3 Required Action A.1 requires that the affected flow path be isolated but does not require testing of the operable valve dependant on the position of the inoperable valve. ITS 3.6.3 Required Action A.1 will delete the CTS requirement to test the other valve and to isolate using only the Operable valve, and adopt the NUREG 3.6.3 requirements to isolate the flow path by isolating the penetration. This isolation can be made with a closed and deactivated automatic valve, a closed manual valve, a blind flange, or a check valve with the flow secured. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L17 CTS 3.6.6 provides 24 hours to be in cold shutdown (ITS MODE 5) upon failure to restore an inoperable reactor building (RB) isolation valve or close the other valve, and does not require entering MODE 3 in any specific time. NUREG 3.6.3 Required Action (RA) E.2 provides a less restrictive 36 hours to be in MODE 5 from a similar Condition. However, NUREG 3.6.3 Required Action E.1 is also included and adds a more restrictive "be in MODE 3 [in] 6 hours." These shutdown actions are consistent with the shutdown sequence and times provided throughout the ITS, and are revised here for consistency. The NUREG 3.6.3 RA E.1 and E.2 requirements will be renumbered and adopted as ITS 3.6.3 Required Actions D.1 and D.2. These Completion Times are consistent with NUREG-1430.

L18 CTS 4.4.1.4 describes Isolation Valve Function Tests as being performed on "remotely operated" reactor building (RB) isolation valves. The "remotely operated" set of isolation valves are considered administratively equivalent to the set described as "each power operated and each automatic" isolation valve. However, NUREG Surveillance Requirement (SR) 3.6.3.5, which has been renumbered for adoption as ITS SR 3.6.3.3, has been modified by TSTF-46 to remove the valves identified as power operated. This is because there are valves credited as RB isolation valves which are power operated that do not receive an RB isolation signal. These power operated valves do not have an isolation time assumed in the accident analysis since they require operator action. Therefore, deleting a reference to power operated isolation valve time testing reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions of the accident analysis. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L19 CTS 3.3.1(I) and CTS 3.3.4(D) require that the engineered safety features valves for the reactor building (RB) spray system, RB cooling system and spray additive system be OPERABLE or locked in the Engineered Safeguards (ES) position whenever the associated system or component was required to be OPERABLE. ITS LCOs 3.6.5 and 3.6.6 will retain these requirements as a condition of system OPERABILITY. However, NUREG-1430 and the ITS allow the ES valves to be verified OPERABLE by actuation to the correct position or by being locked, sealed or otherwise secured in position. The expanded options for administratively controlling valve position will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L20 CTS 3.6.5 requires that the reactor building (RB) manual isolation valves that are required to be closed, be confirmed closed and locked. ITS SR 3.6.3.2 and SR 3.6.3.3 require that the outside and inside reactor building manual isolation valves be verified closed unless locked, sealed or otherwise secured in position. This relaxes the requirement that all valves that are required to be closed be locked closed. Furthermore, the position verification requirements for valves that are closed and locked are allowed to be administratively controlled outside of Technical Specification requirements. The expanded options for controlling and verifying valve position will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

LESS RESTRICTIVE—ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process that are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Program in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

CTS Location	New Location
1.7 a. & e.	B3.6.1 LCO
1.7 a. & b.	B3.6.2 LCO
1.7 c.	B3.6.3 LCO
3.3.4 (A)	B3.6.5 BKG
3.3.4 (D)	B3.6.5 BKG & B3.6.6 BKG
3.14.1	B3.6.7 BKG
3.23.1	B3.6.3 BKG
4.5.2.1.1 (b)	B3.6.5 SR 3.6.5.8
4.5.2.1.2(a)(2)	B3.6.5 BKG & SR 3.6.5.3
4.5.2.1.2(c)(3)	B3.6.5 SR 3.6.5.7
4.12.1 a.	B3.6.7 SR 3.6.7.1
4.12.1 b.2.	B3.6.7 SR 3.6.7.2
4.12.1 b.3.	B3.6.7 SR 3.6.7.3
4.26.1	B3.6.3 SR 3.6.3.1

LA2 This information has been moved to the Technical Requirements Manual (TRM). This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM will be controlled by 10 CFR 50.59 and 10 CFR 50.71, as applicable. This change is consistent with NUREG-1430.

CTS Location	New Location
4.12.1.b.1	TRM

LA3 This information has been moved to a licensee controlled document such as the Reactor Building Leakage Rate Testing Program (RBLRTP), In-Service Testing (IST),and plant procedures, etc. This information provides details of the method of implementation that are not directly pertinent to the actual requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The details relocated to the RBLRTP, RBTSP, and IST will be controlled by 10 CFR 50.59. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
4.4.1.4	IST
4.5.2.1.1 (c)	IST
4.5.2.2.1 & 4.5.2.2.2	IST

3.6.1
3.6.2
3.6.3

1.7 ~~REACTOR BUILDING~~ A1

~~Reactor building integrity exists when the following conditions are satisfied:~~ A18

3.6.1 LCO a. The equipment hatch ~~is closed and sealed and both doors of~~ is OPERABLE the LA1 Bases

3.6.2 LCO personnel lock and emergency lock ~~are closed and sealed, or b. below~~ are OPERABLE A18 LA1 Bases

3.6.2 LCO b. ~~At least one door on each of the~~ personnel lock and emergency lock ~~is closed and sealed~~ are OPERABLE during personnel access or repair. A18

3.6.3 LCO
3.6.3 ACT. Note 1 c. All non-automatic reactor building isolation valves ~~and blind flanges~~ manual valves, check valves, A19 ~~are closed as required~~ are OPERABLE A18 LA1 Bases

3.6.3 LCO d. All automatic reactor building isolation valves are operable or deactivated in the closed position.

SR 3.6.1.1 e. The reactor building leakage ~~determined at the last testing interval satisfies Specification 4.4.1.~~ is in accordance with the RBLRTP A20 LA1 Bases

<Later> (1.0) — Later

~~1.8 FIRE SUPPRESSION WATER SYSTEM~~

~~The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device on each sprinkler system.~~

~~1.9 STAGGERED TEST BASIS~~

~~A staggered test basis shall consist of:~~

~~a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.~~

~~b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.~~

3.6.5

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTO BUILDING SPRAY SYSTEMS

~~Applicability~~

~~Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~ A1

~~Objectivity~~

~~To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor buildi spray systems.~~

Specification

3.6.5 LCO
3.6.5 Appl.
& ⟨Later⟩ (3.3D, 3.5, 3.7)
3.6.5 LCO Note
3.6.5 LCO Note

3.3.1 The following equipment shall be operable in MODES 1,2,3 & 4 ~~whenever containment integrity is established as required by Specification 3.6.1:~~ M1 & Later

(A) One reactor building spray pump and its associated spray nozzle header.

(B) One train of reactor building emergency cooling.

⟨Later⟩ (3.7) ~~(C) Two out of three service water pumps shall be operable, powe from independent essential buses, to provide redundant and independent flow paths.~~ Later

⟨Later⟩ (3.5) ~~(D) Two engineered safety feature actuated Low Pressure Injectio (LPI) pumps shall be operable.~~ Later

⟨Later⟩ (3.7) ~~(E) Both low pressure injection coolers and their cooling water supplies shall be operable.~~ Later

⟨Later⟩ (3.3D) ~~(F) Two Borated Water Storage Tank (BWST) level instrument chanr shall be operable.~~ Later

⟨Later⟩ (3.5) ~~(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.~~ Later

~~(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.~~

3.6.5
3.6.6

3.6.5 LCO & <LATER> (3.5, 3.7)

(I) The engineered safety features valves associated with each of the above systems shall be operable or locked , Sealed, or otherwise secured in the ES position. L19 & LATER

<LATER> (3.5)

3.3.2 ~~In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350F and irradiated fuel is in the core:~~ LATER

~~(A) Two out of three high pressure injection (makeup) pumps shall be maintained operable, powered from independent essential buses, to provide redundant and independent flow paths.~~

~~(B) Engineered safety features valves associated with 3.3.2.a above shall be operable or locked in the ES position.~~

3.3.3 ~~In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:~~

~~(A) The two core flooding tanks shall each contain an indicated minimum of 13 ± 0.4 feet (1040 ± 30 ft³) of borated water at 600 ± 25 psig.~~

~~(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.~~

~~(C) The electrically operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.~~

~~(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.~~

3.3.4 The reactor shall not be ~~made critical~~ in MODE 1 or 2 unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable. A1

3.6.5 LCO & Appl.
3.6.6 LCO & Appl.

(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. ~~The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.~~ LA1 Bases M7

SR 3.6.6.2
SR 3.6.6.3

(B) Verify once each 184 days that The sodium hydroxide tank ~~shall~~ contains a volume of ≥9,000 gallons of sodium hydroxide solution at a concentration >5.0 wt% and <16.5 wt%.

3.6.6 LCO

(C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be ~~locked~~ open. L14

<Add SR 3.6.6.1 & SR 3.6.6.4> M10

A21

3.6.5
3.6.6

3.6.5 LCO
3.6.6 LCO
⟨Later⟩ (3.5, 3.7)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, ⟨sealed, or otherwise secured⟩ in the ES position. & LATER L19

3.3.5 ~~Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~ L3 Later

⟨Later⟩ (3.5, 3.7)

<LATER> (3.5, 3.7) systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance. & LATER L3

<LATER> (3.3D, 3.5, 3.7) LATER

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and, if not corrected, in cold shutdown condition within an additional 72 hours. 6 M18 MODE 3 A1 L6 L15

3.6.5 Cond. A
3.6.5 Cond. D
Restore in 72 hrs or

3.3.7 Exceptions to 3.3.6 shall be as follows:

<LATER> (3.3D) (A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable. LATER

<LATER> (3.5) (B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure of level) shall be operable. LATER

A6

3.6.5 Cond. B (C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, A1 MODE 3

3.6.5 Cond. D restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. L5

A6

3.6.5 Cond. C (D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, A1 MODE 3

3.6.5 Cond. D restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above

3.6.5 Cond. B required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. L5

<Add 3.6.5 RA A.1 & B.1 CT of 10 days> M8

See Pg. 38-1

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance. LATER

<LATER> (3.3D, 3.5, 3.7)

3.3.6 If the conditions of Specifications 3.3.1, ~~3.3.2, 3.3.3,~~ 3.3.4 and 3.3.5 cannot be met ~~except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and,~~ if not corrected, in ~~cold shutdown~~ condition within an additional ~~72~~ hours. 36 In 36 hours MODE 5 A7 A1 A10 M19

3.3.7 Exceptions to 3.3.6 shall be as follows:

See Pg. 38-1

(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.

(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure of level) shall be operable.

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

See Pg. 38-1

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.

⟨LATER⟩ (3.3D, 3.5, 3.7) LATER

3.6.6 Cond.A 3.3.6 If the conditions of Specifications 3.3.1, ~~3.3.2, 3.3.3,~~ 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, ~~reactor shutdown shall be initiated and~~ the reactor shall be in ~~hot shutdown~~ condition within ~~36~~ 6 hours~~, and, if not corrected, in cold shutdown condition within an additional 72 hours.~~ M18 MODE 3 A1 L6

3.6.6 Cond.B restore in 72 hours or L15

3.3.7 Exceptions to 3.3.6 shall be as follows:

(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.

See Pg. 38-1

(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure of level) shall be operable.

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

A14

3.6.5 Cond.B (E) If the conditions of Specification 3.3.4(A) cannot be met because
3.6.5 Cond.A one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least ~~hot shutdown~~ MODE 3 within the next 6 hours ~~and in~~
3.6.5 Cond. D ~~cold shutdown within the following 30 hours~~. Restore the inoperable reactor building emergency cooling train to operable status within 7 days of initial loss or be in at least ~~hot shutdown~~ MODE 3 within the next 6 hours ~~and in cold shutdown within the following 30 hours~~.

A1 L5 A1 L5

⟨Add 3.6.5 Cond. G⟩ M20

Bases

A2

~~The requirements of Specification 3.3.1 assure that below 350°F, adequate long term core cooling is provided. Two low pressure injection pumps are specified. However, only one is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident.~~

~~The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirements assure that the required post-accident components are available for both reactor building emergency cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operable.~~

~~A train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.~~

~~The borated water storage tank is used for three purposes:~~

~~(A) As a supply of borated water for accident conditions.~~

~~(B) As an alternate supply of borated water for reaching cold shutdown.(2)~~

~~(C) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)~~

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent Δk/k subcritical at 70°F without any control rods in the core. A2

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

~~**REFERENCES**~~

~~(1) FSAR, Section 14.2.5~~
~~(2) FSAR, Section 3.2~~
~~(3) FSAR, Section 9.5.2~~
~~(4) FSAR, Section 9.3.1~~
~~(5) FSAR, Section 6.3~~
~~(6) ANO Calculation 91-E-0019-01~~

A2

<CTS INSERT 54A>

<Add 3.6.2 ACTIONS Notes 1, 2, & 3> A8

<Add 3.6.2 Condition A Note 1> A9

<Add 3.6.2 Required Action A.2> M11

<Add 3.6.2 Required Action A.3 & Note> M11

<Add 3.6.2 Condition B with 2 Notes, B.1, B.2 & B.3> M12

<Add 3.6.2 Condition C with RA's C.1, C.2, C.3, & CT's> M12

<Add SR 3.6.3.1 Note & SR 3.6.3.2 Note> L12

<Add SR 3.6.4.1> M13

3.6.1
3.6.2
3.6.3
3.6.4

3.6 REACTOR BUILDING

~~**Applicability**~~ (A1)

~~Applies to the operability of the reactor building.~~

~~**Objective**~~

~~To assure reactor building operability.~~

Specification

3.6.1 LCO
3.6.1 APPL
3.6.2 APPL
3.6.3 APPL

3.6.1 The reactor building shall be operable ~~whenever all three (3) of the following conditions exist:~~ in MODES 1,2,3,4 (M1)

~~a. Reactor coolant pressure is 300 psig or greater.~~

~~b. Reactor coolant temperature is 200°F or greater.~~

~~c. Nuclear fuel is in the core.~~

3.6.2 Act Note 3
3.6.1 RA A.1/B.1/B.2
3.6.2 RA A.1/D.1/D.2
3.6.3 RA D.1/D.2

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least ~~Hot Standby~~ MODE 3 within the next 6 hours and in ~~Cold Shutdown~~ MODE 5 within the following 30 hours. (M2) (M22) (A1)

<LATER (3.9)>

~~3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable.~~ LATER

~~3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% Δk/k shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable.~~ (A3)

3.6.4 LCO
3.6.4 APPL
3.6.4 COND A
3.6.4 RA B.1/B.2

3.6.4 ~~The reactor shall not be taken critical or remain critical~~ MODES 1,2,3,4 if the reactor building internal pressure exceeds 3.0 psig or ~~a vacuum of 5.6 inches Hg~~ -1.0 psig. ~~With the reactor critical~~, restore the containment pressure to within its limits within one hour or be in at least ~~Hot Standby~~ MODE 3 within the next 6 hours and in ~~Cold Shutdown~~ MODE 5 within the following 30 hours. (M3) (M14) (M2) (A1)

3.6.5 ~~Prior to criticality following a refueling shutdown~~, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed ~~and locked~~ as required. if not locked, sealed, secured ~~The provisions of Specification 3.0.3 are not applicable.~~ (L20) (M16)

SR 3.6.3.2: (for outside RB): Every 31 days
SR 3.6.3.3: (for inside RB): Prior to entering MODE 4 from MODE 5 if not performed in previous 92 days (M4)

<CTS INSERT 54A>

<CTS INSERT 54A>

<Add 3.6.2 ACTIONS Notes 1, 2, & 3> A8

<Add 3.6.2 Condition A Note 1> A9

<Add 3.6.2 Required Action A.2> M11

<Add 3.6.2 Required Action A.3 & Note> M11

<Add 3.6.2 Condition B with 2 Notes, B.1, B.2 & B.3> M12

<Add 3.6.2 Condition C with RA's C.1, C.2, C.3, & CT's> M12

<Add SR 3.6.3.2 Note & SR 3.6.3.3 Note> L12

<Add SR 3.6.4.1> M13

<ADD: 3.6.3 ACTIONS NOTES 2 & 3> M5
<ADD: 3.6.3 COND A NOTE> A5
<ADD: 3.6.3 RA A.2, & NOTES & CTs> M5
<ADD: 3.6.3 COND B & NOTE>
<ADD: 3.6.3 ACTION C & NOTES>

3.6.3

MODES 1,2,3,4 M3

3.6.3 APPL 3.6.6 If, ~~while the reactor is critical,~~ a reactor building isolation valve is determined to be inoperable in a position other than the closed position, the other reactor building isolation valve (except for check valves) in the line shall be tested to insure operability. If the inoperable valve is not restored within 48 hours, the reactor shall be brought to the cold shutdown condition within an additional 24 hours or the operable valve will be closed.

L16

A1

3.6.3 RA A.1

3.6.3 RA D.2

3.6.3 RA D.1

MODE 5 A1

MODE 3: 6 hrs
MODE 5: 36 hrs

L17

~~**Bases**~~

~~Included in reactor building operability are both the reactor building integrity as defined in Specification 1.7 and the reactor building structural integrity. Structural integrity limitations as described in the ANO Containment Inspection Program ensure the reactor building will be maintained comparable to the original design standards throughout the facility life span. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code as set forth in 10 CFR 50.55a(g)(6)(ii)(B). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.~~

~~The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there will be no pressure buildup in the reactor building if the reactor coolant system ruptures.~~

~~The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.~~

~~The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110°F and the building is subsequently cooled to an internal temperature of less than 50°F.~~

~~When reactor building integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.~~

~~**REFERENCE**~~

~~FSAR, Section 5.~~

A2

3.14 HYDROGEN RECOMBINERS

Applicability — A1

Applies to the operating status of the hydrogen recombiner systems.

Objective

To ensure that the hydrogen recombiner systems will perform within acceptable levels of efficiency and reliability.

LA1 Bases

Specification

3.6.7 LCO Appl. 3.14.1 Two independent hydrogen recombiner systems shall be operable ~~whenever reactor building integrity is required~~. in MODES 1&2 — L7

A1

RA.A.1 3.14.2 With one hydrogen recombiner system inoperable, restore the inoperable system to operable status within 30 days or the reactor

RA.B.1 shall be placed in the hot shutdown condition within the next 6 hours. MODE 3 — A1

<Later (3.3D)> 3.14.3 Hydrogen concentration instruments shall be operable. — Later

3.14.4 With one of two hydrogen concentration instruments inoperable restore the inoperable analyzer to OPERABLE status within 30 days or be in at least hot shutdown within the next 6 hours.

Bases — A2

The hydrogen recombiner systems are designed to operate as necessary to limit the hydrogen concentration in the reactor building following a Loss of Coolant Accident.

The system is composed of two redundant 100% capacity Internal Electrical Hydrogen Recombiners, manufactured by Westinghouse.

<Add 3.6.7 RA A.1 Note> — L13

3.23 REACTOR BUILDING PURGE VALVES

APPLICABILITY

This specification applies to the reactor building purge supply and exhaust isolation valves. A1

OBJECTIVE

To specify that reactor building isolation/purge valves be closed whenever containment integrity is required by TS 3.6.1. LA1 Bases

SPECIFICATION

3.6.3 LCO Appl. ACT. Note 1 3.23.1 The reactor building purge supply and exhaust isolation valves shall be closed and handswitch keys removed whenever containment integrity is required by TS 3.6.1. in MODES 1,2,3,4 M1

BASES

The reactor building supply and exhaust isolation valves are required to be closed during normal plant operation in order to ensure reactor building integrity. Purging is allowed only when containment integrity is not required by TS 3.6.1. A2

3.6.3

Table 4.1-2
Minimum Equipment Test Frequency

	Item	Test	Frequency	
⟨Later⟩ (3.1)	1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown	LATER
	2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions	
⟨Later⟩ (3.4B)	3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month	LATER
⟨Later⟩ (3.7)	4. Main Steam Safety Valves	Setpoint	Four Valves Every 18 Mon	LATER
	5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown	R TRM
⟨Later⟩ (3.4B)	6a. Reactor Coolant System Leakage	Evaluate	Daily	LATER
	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	
	7. Emergency-powered Pressurizer Heaters	Power availability	Daily	
		Heater capacity functional test	Every 18 Months	A16
SR 3.6.3.5	8. Reactor Building Isolation Trip	Functioning Verify auto. RB iso. valves actuate on act. or Sim. Signal	Every 18 Months	
⟨Later⟩ (3.7)	9. Service Water Systems	Functioning	Every 18 Months	LATER
	10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool	R TRM

⟨Later⟩ (3.1) 1/ Same as tests listed in Section 4.7 — Later

Notes:

⟨Later⟩ (3.4B) — LATER

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
⟨Later⟩ (3.4B) 1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
⟨LATER⟩ (3.1)	d. Dissolved Gases	d. Weekly (7)	LATER
⟨LATER⟩ (3.4A)	e. Chemistry (Cl, F, and/O_2)	e. 3 times/week (8)	LATER
⟨Later⟩ (3.9)	f. Boron Concentration	f. 3 times/week	LATER & R TRM
⟨Later⟩ (3.4B)	g. Radiochemical Analysis for Ē Determination (2) (4)	g. Monthly (7)	LATER
⟨Later⟩ (3.5) 2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	
⟨Later⟩ (3.7) 4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LATER
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10	R TRM
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	M 7
SR 3.6.6.3 6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	~~Quarterly and after each makeup~~ every 184 days	L 9

⟨Later⟩ (3.4B)

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

3.6.1
3.6.2
3.6.3

3.6 4.4 REACTOR BUILDING

4.4.1 Reactor Building Leakage Tests

~~Applicability~~

~~Applies to the reactor building.~~

~~Objective~~

~~To verify that leakage from the reactor building is maintained within allowable limits.~~ A1

Specification

SR 3.6.1.1 4.4.1.1 Integrated leakage rate tests shall be conducted and visual inspections performed in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.1.1 Deleted~~
~~4.4.1.1.2 Deleted~~
~~4.4.1.1.3 Deleted~~ A1

SR 3.6.1.1 4.4.1.1.4 Integrated leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.1.5 Deleted~~
~~4.4.1.1.6 Deleted~~
~~4.4.1.1.7 Deleted~~ A1

SR 3.6.1.1
SR 3.6.2.1 4.4.1.2 Local leakage rate tests shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.2.1 Deleted~~
~~4.4.1.2.2 Deleted~~
~~4.4.1.2.3 Deleted~~
~~4.4.1.2.4 Deleted~~ A1

SR 3.6.1.1
SR 3.6.2.1 4.4.1.2.5 Local leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program. A1

~~4.4.1.3 Deleted~~ LA3 IST

SR 3.6.3.4 4.4.1.4 Isolation Valve Functional Tests L18

~~Every three months,~~ ~~remotely operated~~ Automatic reactor building isolation valves shall be stroked and timed to the position required to fulfill their safety function ~~unless such operation is not practical during plant operation. The latter valves shall be tested once every 18 months.~~ A17 LA3 IST

~~4.4.1.5 Deleted~~ A1

⟨Add SR 3.6.2.1 Notes⟩ M11

⟨Add SR 3.6.2.2⟩ M12

3.6.1

A2

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285°F.

The peak calculated reactor building pressure for the design basis loss of coolant accident, P_a, is 54 psig. The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

The reactor building will be periodically leakage tested in accordance with the Reactor Building Leakage Rate Testing Program. These periodic testing requirements verify the reactor building leakage rate does not exceed the assumptions used in the safety analysis. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

REFERENCE

(1) FSAR, Sections 5 and 13.

4.5.2 Reactor Building Cooling Systems

Applicability (A1)

Applies to testing of the reactor building emergency cooling systems.

Objective

To verify that the reactor building emergency cooling systems are operable.

Specification

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

SR 3.6.5.5 SR 3.6.5.6 (a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the reactor building spray system (except for reactor building inlet valves to prevent water entering nozzles).

SR 3.6.5.8 (b) Station compressed air or smoke will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every five years. 10 (LA1 Bases) (L10)

(c) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly. (LA3 IST)

4.5.2.1.2 Reactor Building Cooling System 31 (L11)

SR 3.6.5.3 (a) At least once per 14 days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

SR 3.6.5.3 (1) Verifying a service water flow rate of ≥ 1200 gpm to each train of the reactor building emergency cooling.[1] (A12)

(2) Addition of a biocide to the service water during the surveillance in 4.5.2.1.2.a.1 above, whenever service water temperature is between 60F and 80F. (LA1 Bases)

SR 3.6.5.2 (b) At least once per 31 days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) Starting (unless already operating) each operational cooling fan from the control room.

(A12)

[1] Surveillance Requirement 4.5.2.1.2(a)(1) will not be performed on the green train of the reactor building emergency cooling system until cooling fan VSF-1D is repaired and the green train is returned to normal configuration. This note will remain in effect until July 14, 1995.

3.6.5

(2) Verifying that each operational cooling fan operates for at least 15 minutes.

SR 3.6.5.7 & LATER (3.7) (c) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

SR 3.6.5.7 (1) A test signal will be applied to actuate the reactor building emergency cooling operation.

<Later> (3.7) ~~(2) Verification of the engineered safety features function of the service water system which supplies the reactor building emergency coolers shall be made to demonstrate operability of the coolers.~~ Later

SR 3.6.5.7 &<Later> (3.7) (3) The test will be considered satisfactory if ~~control board indication verifies that~~ all components have responded to the actuation signal properly. LA1 Bases & Later

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

SR 3.6.5.4 ~~At intervals not to exceed 3 months the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.~~ LA3 IST

4.5.2.2.2 Valves

~~At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building emergency cooling system and each engineered safety features valve associated with reactor building emergency cooling in the service water system shall be tested to verify that it is operable.~~ LA3 IST

<LATER> (3.7) LATER

&<LATER> (3.7) LA3 IST & LATER

<Add SR 3.6.5.1> M15

Bases

~~The reactor building emergency cooling system and reactor building spray system are redundant to each other in providing post-accident cooling of the reactor building atmosphere to prevent the building pressure from exceeding the design pressure. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building emergency cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable engineered safeguard equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.~~ A2

Addition of a biocide to service water is performed during reactor building emergency cooler surveillance to prevent buildup of Asian clams in the coolers when service water is pumped through the cooling coils. This is performed when service water temperature is between 60F and 80F since in this water temperature range Asian clams can spawn and produce larva which could pass through service water system strainers.

A2

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or smoke can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the reactor building emergency cooling system are arranged so that they can be visually inspected. The cooling fans and coils and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.

As the reactor building fans are normally operating, starting for testing is unnecessary for those verified to be operating.

Reference

FSAR, Section 6

4.12 HYDROGEN RECOMBINERS SURVEILLANCE

~~Applicability~~

~~Applies to the surveillance of the hydrogen recombiner systems.~~

~~Objective~~

~~To verify an acceptable level of efficiency and operability of the hydrogen recombiner systems.~~

A1

SR 3.6.7.1

Specification

4.12.1 Each hydrogen recombiner system shall be demonstrated OPERABLE: L8

a. At least once per 6 months 18 Months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 KW. LA1 Bases

SR 3.6.7.2
SR 3.6.7.3

b. At least once per 18 months by:

1. ~~Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,~~ LA2 TRM

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure ~~(i.e., loose wiring or structural connections, deposits of foreign materials, etc.)~~, and AL3 LA1 Bases

3. Verifying the integrity of the heater electrical circuits by performing a resistance to ground test following the above required functional test. ~~The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.~~ LA1 Bases

<Later> (3.3D) ~~4.12.2 Hydrogen concentration instruments shall be calibrated once every 18 months with proper consideration to moisture effect.~~ LATER

~~Bases~~

~~The OPERABILITY of the recombiners for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the~~ A2

3.6.7

~~expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. The hydrogen recombiner systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following LOCA", Rev. 2, November, 1978.~~

A2

3.6.3

4.26 REACTOR BUILDING PURGE VALVES

Applicability

This specification applies to the reactor building purge supply and exhaust isolation valves. A1

Objective

To assure reactor building integrity.

LA1 Bases

Specification

M1

SR 3.6.3.1 4.26.1 The reactor building purge supply and exhaust isolation valves shall be determined closed at least once per 31 days when containment integrity is required by TS 3.6.1. during MODES 1, 2, 3 and 4

<Later> (5.0) 4.26.2 Prior to exceeding conditions which require establishment of reactor building integrity per TS 3.6.1, the leak rate of the purge supply and exhaust isolation valves shall be verified to be within acceptable limits per TS 4.4.1, unless the test has been successfully completed within the last three months. LATER

Bases

Determination of reactor building purge valve closure will ensure that reactor building integrity is not unintentionally breached. A2

As a result of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," it was concluded that excess leakage past valve resilient seals is typically caused by severe environmental conditions and/or wear due to use. Recommended leak test frequencies of three months are deemed to be adequate to detect seal degradation of resilient seals.

The three month test need not be conducted with the precision of the Type C 10CFR50, Appendix J criteria, however the test must be sufficient to detect degradation.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ANO-1 ITS SECTION 3.6 : Reactor Building Systems

"L" - Less Restrictive Changes to Requirements:

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

3.6 L1 Not used.

3.6 L2 Not used.

3.6 L3

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

Availability of the reactor building cooling system and the reactor building spray system (with the sodium hydroxide system) is needed to mitigate several previously analyzed accidents, however, the systems are not accident initiators and therefore the change doesn't affect the probability of an accident. Elimination of the need to demonstrate the OPERABILITY of one redundant system train in order to perform maintenance on the other system train doesn't affect the consequences of an accident as the redundant trains are 100% capacity and performing maintenance on one doesn't impact the availability of the other.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change doesn't necessitate a physical alteration of the unit (no new or different kinds of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still ensure the required reactor building cooling and spray system capacity is available when required or prompt and appropriate compensatory actions will be taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety because removing the need to demonstrate the OPERABILITY of an already OPERABLE, redundant system train prior to maintenance has no impact on the ability of the system to function in the maintenance configuration. Therefore, this change does not affect the margin of safety, especially for system trains which are also 100% capacity.

3.6 L4 Not used.

3.6 L5

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The reactor building (RB) cooling system and the RB spray system (with the sodium hydroxide system) are needed to mitigate previously analyzed accidents which are assumed to start with the reactor critical at full power in MODE 1. Since the start conditions of an accident are not affected by whether the unit is taken to MODE 3 or MODE 5 in a shutdown, this change doesn't affect the probability of an accident. Furthermore, the Required Action to proceed to MODE 3 rather than MODE 5 does not significantly affect the consequences of an accident. Therefore this change of the Required Action does not significantly increase the consequences of any previously analyzed accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed), a change to a previous analysis or changes in parameters governing normal unit operation. The proposed change will still ensure prompt and appropriate compensatory actions are taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since placing the unit in MODE 3 places the reactor in a sub-critical state. Additionally, during MODES 3 and 4, the reactor coolant is reduced to a temperature and pressure significantly below operating conditions at power and, therefore, no core damage is anticipated from a LOCA initiated during shutdown. Since a MODE 3 or 4 event is not analyzed in the ANO safety analysis, the reactor building OPERABILITY is maintained during MODES 3 and 4 to ensure that the offsite radiation exposure of 10CFR100 are not exceeded.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L6

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The reactor building (RB) spray system and sodium hydroxide system are needed to mitigate previously analyzed accidents which are assumed to start with the reactor critical at full power in MODE 1. Since the start conditions of an accident are not affected by whether the unit is taken to MODE 3 or MODE 5 in a shutdown, this change doesn't affect the probability of an accident. Furthermore, the Required Action to proceed to MODE 3 rather than MODE 5 does not significantly affect the consequences of an accident. Therefore this change of the Required Action does not significantly increase the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed), a change to a previous analysis or changes in parameters governing normal unit operation. The proposed change will still ensure prompt and appropriate compensatory actions are taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since placing the unit in MODE 3 places the reactor in a sub-critical state. Additionally, each of the trains of RB spray and sodium hydroxide (two independent trains) is 100% capacity and whether all or some portion of more than one train of their combined capacity is available, is non-significant to the margin-of-safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L7

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The hydrogen recombiners are available in the reactor building to mitigate hydrogen buildup in the reactor building atmosphere after an accident. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Changing the Applicability of this equipment to MODES 1 and 2 has little or no effect on the consequences of an accident previously evaluated since the Loss of Coolant Accident (LOCA) Design Basis Accident, which this equipment mitigates, assumes accident initiation during MODE 1.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still ensure prompt and appropriate compensatory actions are taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since the applicable MODES for the hydrogen recombiners include MODE 1 which is the starting mode for the LOCA analyses as well as the MODE from which the maximum hydrogen concentration could be produced.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L8

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The hydrogen recombiners are not an accident initiator and no hardware changes are proposed, therefore, the extension of the system function test surveillance Frequency from 6 months to 18 months doesn't affect the probability of a previously analyzed accident. The surveillance Frequency has no impact on the availability expectations for this equipment, especially since each train is 100% capacity, and indicates little if any impact on the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still ensure the availability of the hydrogen recombiners for prompt and appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Extending the surveillance Frequency doesn't affect the actual availability of the equipment. Also, each train of the hydrogen recombiners has 100% of the required hydrogen concentration reduction capacity which provides significant redundancy and indicates that the margin of safety is not significantly reduced.

<u>3.6 L9</u>

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The sodium hydroxide (NaOH) concentration is not an accident initiator and there are no hardware changes proposed, therefore the extension of the sampling Frequency and the elimination of a requirement for testing after each makeup doesn't affect the probability of an accident. There is a low probability of uncontrolled changes to the sodium hydroxide concentration since the tank is normally isolated from a makeup source and there is tank volume indication in the control room. These controls are adequate to maintain NaOH concentration at a level which will not cause an adverse impact and will not affect the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the availability of sodium hydroxide in sufficient concentration for appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Extending the surveillance Frequency doesn't affect the actual concentration of the sodium hydroxide but rather the potential that the concentration might be altered without being noticed. However, the controls on the sodium hydroxide concentration are such that changes in concentration would be noticed and corrective action taken. Therefore, extension of the surveillance Frequency does not involve a significant reduction in a margin of safety.

3.6 L10

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The reactor building spray headers and nozzles are not an accident initiator and there are no hardware changes proposed, therefore the extension of flow path testing from 5 years to 10 years doesn't affect the probability of an accident. Testing of the reactor building spray headers and nozzles for obstruction and function has indicated good performance and supports the conclusion that the availability of this passive component is not affected by changing the surveillance Frequency. The lack of impact on the ability of the spray headers and nozzles to function as intended indicates that there is no significant impact on the consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the availability of the reactor building spray headers and nozzles for appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since extending the surveillance Frequency will introduce only a small increase in the potential that a nozzle or header could be obstructed or degraded between surveillances. Previous testing indicates good availability performance of the reactor building spray headers and nozzles and supports the conclusion that there is no significant reduction in a margin of safety. Furthermore, it should be recognized that each train of the RB spray system is independent and redundant. This level of system redundancy also supports the conclusion that there is no significant reduction in a margin of safety from the extension of surveillance Frequency.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L11

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The reactor building (RB) cooling train service water flow rate test frequency is not an accident initiator and there are no hardware changes proposed, therefore the extension of the flow rate test Frequency doesn't affect the probability of an accident. Whether the consequences of a previously analyzed accident are affected is dependent on the availability of adequate service water flow to the RB coolers. The surveillance Frequency of 31 days is consistent with the test Frequency of other components in the RB cooling system and is appropriate for a system which is considered reliable. The reliability of adequate service water flow to the RB coolers indicates that extending the service water flow test Frequency from 14 days to 31 days will not affect the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the availability of adequate reactor building (RB) cooling system service water flow for appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since extending the surveillance Frequency will introduce only a small increase in the potential that the reactor building (RB) cooling system service water flow could be degraded between surveillances. Previous testing indicates good availability performance of the RB cooling system service water flow. Furthermore, it should be recognized that each train of the RB cooling system, including the service water supply, is independent and 100% capacity. This level of system redundancy also supports the conclusion that there is no significant reduction in a margin of safety from the extension of surveillance Frequency.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L12

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The adoption of an alternate method for position verification of valves in high radiation areas is not an accident initiator and there are no hardware changes proposed. Therefore, the introduction of an option to validate valve position by administrative means does not affect the probability of an accident. The consequences of an accident previously evaluated could only be affected if the valves of interest are out of proper position. The administrative control of valves in high radiation areas is considered adequate to maintain the proper position of this valve population. This conclusion is based on the limited accessibility of valves in this population, the strength of the ALARA program and the effectiveness of other administrative control programs. The adoption of an alternate method for position verification of valves in high radiation areas is not considered to cause a significant increase in the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the proper positioning of valves to provide compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since just altering the method of verifying correct position for the limited set of valves located in a high radiation area has a non-significant impact on the OPERABILITY of the valves in question.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>3.6 L13</u>

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The hydrogen recombiners are available in the reactor building to mitigate hydrogen buildup in the reactor building atmosphere after an accident. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Adopting a Note which makes LCO 3.0.4 not applicable while one of the hydrogen recombiners is inoperable, has no impact on the consequences of a previously evaluated accident. This conclusion is based on the availability of the other, 100% capacity, hydrogen recombiner which would be available to mitigate an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since each hydrogen recombiner is 100% capacity and the probability of a LOCA occurring while one of the hydrogen recombiners is inoperable is very small.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L14

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

Sodium hydroxide (NaOH) is added to the reactor building (RB) spray solution for pH control. There are two independent flow paths, including manual and control valves, which provide gravity flow of the NaOH solution to the RB spray headers. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Deleting a requirement that the manual valves in the sodium hydroxide system be locked open has no impact on the consequences of a previously evaluated accident. Locking these valves open has small impact on the availability of the NaOH system which is virtually eliminated by the Frequency of valve position surveillance which is adopted.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since deleting the requirement to lock open the sodium hydroxide manual valves has minimal impact on the availability of the sodium hydroxide system. Furthermore, valve position surveillance, which is maintained, for the sodium hydroxide system is considered sufficient to provide system availability in the event of an accident.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L15

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The restoration time for an inoperable reactor building spray train or sodium hydroxide system is not an accident initiator and, since no hardware changes are proposed, the change doesn't affect the probability of an accident. The extension of the restoration time is a non-significant change to the consequences to previously evaluated Design Basis Accidents. This conclusion is based on the small probability that an accident will occur during the extended restoration time.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since the extension of the restoration time has minimal impact on the availability of individual trains of the reactor building (RB) spray system or the sodium hydroxide system. Furthermore, the change to the margin of safety is not impacted since the other trains, which remain OPERABLE, are each 100% capacity.

3.6 L16

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change will delete the requirement to test the redundant, OPERABLE, valve when the first valve is declared inoperable in a two valve reactor building (RB) penetration. Furthermore, the change will relax the limitation the isolation must occur using only the Operable valve; allowing the isolation to be made with a closed and deactivated automatic valve, a closed manual valve, a blind flange, or a check valve with the flow secured. The testing of RB penetrations is not an accident initiator and there are no penetration hardware changes proposed, therefore, the change doesn't affect the probability of an accident. Deleting a requirement that the second valve be tested has no impact on the consequences of a previously evaluated accident. This is because testing the "other" valve has no impact on whether the reactor building can be isolated by the closing of RB isolation valves. The allowed methods of isolating the penetration are only those already included in the plant design and therefore, there will be no impact on the consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions (the isolation valves will be closed) will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since deleting the requirement to test the redundant, OPERABLE, valve when one valve in a two valve reactor building (RB) penetration is declared inoperable, has minimal impact on the availability of the penetration to provide reactor building isolation. Isolating the penetration provides assurance that penetration function is maintained in compliance with design and hence margin of safety is not significantly reduced.

3.6 L17

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

This change will extend the Completion Time for taking the unit to MODE 5 upon failure to restore an inoperable reactor building (RB) isolation valve or isolate the RB penetration flow path. This change and equipment is not an accident initiator and since no hardware changes are proposed, the change doesn't affect the probability of an previously evaluated accident. The extension of the Completion Time is consistent with the time allowed in NUREG-1430 and is furthermore a small extension of the transition period to MODE 5. This change is a non-significant impact on the consequences of an accident since the accident parameters are not affected, only a limited extension of the time the unit would be in MODE 1, 2, 3 or 4.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since no accident parameters are changed.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.6 L18

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

This change will delete the set of "power operated" valves, which do not receive a reactor building isolation signal, from the set of "automatic" valves, which do receive a reactor building isolation signal, for the purpose of valve closure time testing. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. The consequences of a previously evaluated accident are not impacted since the valves included in the accident analysis are part of the set of automatic valves and there is , therefore, no change to the analysis assumptions and conclusions.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since there is no change to the accident scenario; only a change to the surveillance requirements of "power operated" valves which do not receive a reactor building isolation signal.

3.6 L19

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change will introduce the option to lock, seal, or otherwise secure the engineered safeguards (ES) valves for the reactor building (RB) spray system and the RB cooling system when RB OPERABILITY is required. Before this change, the only option was to lock the valves in position. The method of verifying ES valve position is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Expanding the methods available for verifying ES valve position has no impact on the consequences of a previously evaluated accident since the valves of interest are still placed in proper position for their safety function.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since expanding the methods of verifying the RB spray system and RB cooling system ES valves has minimal impact on the availability of the systems and the valves. Furthermore, valve position surveillance, regardless of method of verification, is considered sufficient to provide system availability in the event of an accident.

3.6 L20

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change will introduce the option to close but leave unlocked, seal, or otherwise secure the inside and outside reactor building manual isolation valves when reactor building OPERABILITY is required. Before this change, the only option was to close and lock the valves. Furthermore, if locked, sealed or otherwise secured, the required position verification is allowed to be administratively controlled outside of Technical Specifications. The method and frequency of verifying the inside and outside reactor building manual isolation valve position is not an accident initiator and no hardware changes are proposed; therefore, the change doesn't affect the probability of an accident. Removing the requirement for the administrative control of locking, sealing, or otherwise securing the inside and outside reactor building manual isolation valves in position has no impact on the consequences of a previously evaluated accident since the valves of interest are still placed in proper position for their safety function.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since removing the requirement for the administrative control of locking, sealing, or otherwise securing the inside and outside reactor building manual isolation valves in position has minimal impact on the availability of the systems. Furthermore, the removing the re-verification of the position of valves that have been locked, sealed, or otherwise secured (they were verified to be in position when locked, sealed, or otherwise secured) is does not involve a significant reduction in a margin of safety. The original verification and concurrent administrative controls is considered sufficient to provide assurance of reactor building OPERABILITY in the event of an accident.

1. NUREG Section 3.6 - The title of this section is changed from "Containment" to "Reactor Building" Systems to comply with ANO-1 terminology in the license basis documents. References to "containment" in the NUREG-1430 text are changed to "reactor building," "the reactor building," or the abbreviation "RB" as appropriate for the ITS context. However, marking up the NUREG pages to show these changes introduces significant clutter to the page with little value for the purpose of the markup. Therefore, only one reference to this DOD item will be placed on each page of the NUREG/ITS markup for this section at the first occurrence (usually in the section title area) with subsequent changes on that page not marked or annotated with this DOD number to conserve margin space.

   In the NUREG-1430 3.6 Bases, use of the term "containment" is very frequent and showing the change would affect readability and reduce margin space. In order not to clutter the markup with material of negligible value, only the first occurrence of the change (usually in the section title area) will be shown in the ITS Bases markup and annotated once on each page; therefore, the word "containment" should be read as "reactor building" (or "the reactor building") wherever it occurs in the Bases text. However, since a generic rule usually doesn't apply to every case, some changes will still be marked in the Bases text as appropriate for sentence clarification, unit specific changes, or to provide emphasis. These changes are consistent with current license basis.

2. NUREG SR 3.6.1.1, SR 3.6.2.1 – The ITS is revised to reflect ANO-1 CTS 4.4.1 requirements. These surveillance requirements indicate that the reactor building leakage rate testing is to be performed in accordance with the Reactor Building Leakage Rate Testing Program described in CTS 6.8.4. These CTS requirements were established by Amendment 185 which implemented 10 CFR 50, Appendix J, Option B. In the ITS, the Reactor Building Leakage Rate Testing Program will be covered by ITS Specification 5.5.6. The Bases were revised as appropriate. These changes are consistent with current license basis.

3. NUREG 3.6.2 Bases, Background, - 3rd paragraph is deleted since it does not reflect the ANO-1 design.

4. NUREG SR 3.6.3.3, SR 3.6.3.4 - Incorporated TSTF-45, Rev. 2.

5. NUREG SR 3.6.3.5 (ITS SR 3.6.3.4) – Incorporated TSTF-46, Rev. 1.

6. NUREG 3.6.3 Required Action C.1 Completion Time – Incorporates TSTF-30, Rev. 3.

   NUREG 3.6.3 Bases - The TSTF-30, Rev. 3, reference to Standard Review Plan 6.2.4 in the NUREG 3.6.3 Condition C Bases and the subsequent reference number changes are not adopted as the Standard Review Plan is not consistent with the ANO-1 license basis.

7. The ANO-1 current license basis (CTS 3.23) requires that the reactor building (RB) purge valves be closed and not operated whenever RB integrity is required (that is, above cold shutdown). The current license basis for the RB purge valves will be maintained in the ITS.

   NUREG 3.6.3 ACTIONS Note 1 was modified and NUREG SR 3.6.3.2 was deleted since the ANO-1 RB purge system design features one suction and one discharge train with 24 inch valves in series (one inside and one outside the reactor building) and has no smaller-diameter bypass purge valves or alternate purge flow paths.

   NUREG 3.6.3 ACTIONS Note 4 was deleted since the unique purge valve leakage criteria for which it was primarily intended are not applicable at ANO-1. For other reactor building isolation valves, also with no special leakage requirements, this Note is redundant to the requirements of ITS LCO 3.6.1.

   NUREG 3.6.3 ACTIONS A, B, and D - In accordance with ANO-1 current license basis, the ANO-1 RB purge isolation valves are closed and not operated whenever RB integrity is required. As a result of this requirement, the RB purge valves in the CTS are considered to be "otherwise secured" for the duration of the operating cycle and therefore do not receive special leakage monitoring above cold shutdown. The CTS requirements for the ANO-1 RB purge isolation valves require closing these valves and removing the handswitch key when RB integrity is required and furthermore provides that operation of the purge valves when RB integrity is required is specifically prohibited by CTS 3.23. NUREG 3.6.3, in Conditions A, B, and D, addresses purge valves uniquely with respect to other RB isolation valves assuming that these valves may be operated during periods when the RB must be OPERABLE. Since these valves at ANO-1 are not operated when the RB must be OPERABLE, then their OPERABLE condition for the ITS will be the same as in CTS 3.23.1 which is "closed with the handswitch key removed" and therefore, no different than other RB isolation valves. These CTS requirements will be adopted in the ITS by deleting the purge valve references from NUREG 3.6.3 Condition A and Condition B, by deleting NUREG 3.6.3 Condition D in its entirety, and by deleting the "sealed" term which has no definition for these valves in the CTS.

   Since NUREG 3.6.3 Condition D was deleted, Condition E was renumbered to Condition D for adoption by the ITS.

   NUREG SR 3.6.3.1 is revised for adoption in the ITS to reflect the ANO-1 purge isolation valve configuration and operation in a manner consistent with CTS 4.26.1. NUREG-1430 ASA Bases, SR 3.6.3.2, and SR 3.6.3.8 are not adopted by the ITS since these surveillances address a purge system configuration which is not applicable to ANO-1. NUREG SR 3.6.3.6 is not adopted by the ITS since leakage rate testing for the purge valves at ANO-1 is no different than that done for other reactor building isolation valves that are closed during MODES 1, 2, 3, and 4. The leakage rate testing for the purge valves is done as part of the Reactor Building Leakage Rate Testing Program (RBLRTP) and is addressed in ITS Section 5.0. Since these NUREG SRs are not adopted, the other

NUREG SRs are renumbered as follows: SR 3.6.3.3 to ITS SR 3.6.3.2, SR 3.6.3.4 to ITS SR 3.6.3.3, SR 3.6.3.5 to
ITS SR 3.6.3.4, and SR 3.6.3.7 to ITS SR 3.6.3.5.

The NUREG-1430 3.6.3 Bases were modified as appropriate for ANO-1 specific purge valve changes.

8. NUREG SR 3.6.2.2 – Incorporates TSTF-17, Rev. 2, as modified for the ANO-1 fuel cycle length. The present outage cycle for ANO-1 is approximately 18 months, therefore the Frequency is changed from 184 days to 18 months instead of 24 months as shown in TSTF-17, Rev. 2.

9. The NUREG-1430 title for NUREG LCO 3.6.7, the "Spray Additive System", is adopted for ITS LCO 3.6.6. Adopting this name for the system is considered to emphasize its post-LOCA radioactive iodine-removal function and is consistent with the design intent. However, the CTS terminology for the components of the system is retained in the ITS. Therefore, a specific component reference, for instance in a Surveillance Requirement, to the "sodium hydroxide tank" or "NaOH tank" will not become the "spray additive tank". This change is consistent with NUREG-1430 as well as with the CTS. The ITS 3.6.6 Bases are revised as appropriate for component names.

10. NUREG 3.6.8 Condition B - ANO-1 has no alternate hydrogen control system for use when two hydrogen recombiners are inoperable; therefore, NUREG 3.6.8 Condition B is not applicable and is not adopted in the ITS. The subsequent NUREG 3.6.8 Condition C is renumbered to Condition B for adoption by the ITS. The discussion of when this deleted NUREG Condition could be adopted is retained in the ITS Bases. This change is consistent with current license basis.

11. NUREG 3.6.2 Bases Background - ANO-1 terminology differentiates between the two air locks. The two ANO-1 air locks are similar in function but different in size and mechanism. The NUREG 3.6.2 Bases are revised where appropriate for the different air locks. Only the initial change is annotated for this item on a page in order to save margin space. The remaining changes on a page are marked up but not annotated. This change is consistent with current license basis.

12. NUREG 3.6.5 is not adopted in the ITS since there are no comparable requirements in the ANO-1 CTS for reactor building (RB) air temperature. Administrative monitoring of the ANO-1 RB temperature indicates that the actual RB bulk air temperature is maintained at a conservative value with respect to the design basis bulk air temperature. ANO-1 has completed modifications which address RB bulk average temperature concerns that were recognized in the 1987 timeframe (1CAN088707). The current condition and operating experience for ANO-1 is that there is a significant margin between the design basis accident analysis initial temperature (140°F) and the maximum observed RB bulk air temperature (less than 120°F). This margin, furthermore, is not sustained by additional active components but rather by passive design changes such as improving the insulation on the

RCS components and improvements to the existing RB air distribution systems. Operating experience for ANO-1 indicates that maintenance of a conservative RB temperature is a robust process with an adequate administrative monitoring system in place. This change is consistent with current license basis.

13. NUREG SR 3.6.7.5 is not adopted in the ITS. The ANO-1 Spray Additive System design features gravity flow of the NaOH solution from the NaOH tank to the Borated Water Storage Tank (BWST) headers upstream of the suction of the reactor building (RB) spray pumps. There are no means of directly measuring flow-before-run-out or rate-of-flow of the NaOH solution for the purposes of surveillance. Assurance of introduction of an adequate amount of NaOH during the design basis accident (DBA) was established analytically based on the passive features of the system including BWST and NaOH tank levels and time to run-out, flow characteristics of the associated pumps and valves, and the associated piping configurations. The availability of the active components of the Spray Additive System is verified by other ITS Surveillance Requirements. This change is consistent with current license basis.

14. NUREG 3.6.4 Bases - ITS 3.6.4 Bases discussion for reactor building (RB) pressure is revised to address inclusion of the Emergency Core Cooling System analysis as a contributor to the limits on RB pressure. This Bases discussion is also revised to address the fact that the LOCA pressure margin to design pressure is a feature of the analysis results rather than the analysis input and that the tornado loads have no effect on the reactor building pressure limit values. This change is consistent with current license basis.

15. NUREG 3.6.6 Bases - The ITS 3.6.5 Applicable Safety Analysis Bases discussion of cooling system total delay time was revised to reflect ANO-1 unit specific design and analysis results. The ANO-1 LOCA analysis conservatively assumes that the RB spray and cooling systems do not affect the early stages of the event and that they are capable of functioning in a degraded condition when they are used. The ITS 3.6.5 LCO Bases discussion of the Reactor Building (RB) Spray System and the RB Cooling System was revised to include unit specific details of the ANO-1 configuration. The ITS 3.6.5 Bases, in general, are revised to describe the significant differences of the ANO-1 RB cooling system configuration from that depicted in NUREG-1430. These changes are consistent with current license basis.

16. NUREG 3.6.7 Bases – ITS 3.6.6 Bases for the Spray Additive System were revised to provide specific design details of the Spray Additive System and components. These details include: gravity flow of sodium hydroxide and mixing, absence of capability to directly measure sodium hydroxide solution flow and details of the sodium hydroxide tank level indication. These changes are consistent with the current license basis.

17. NUREG 3.6.8 Bases - The ITS 3.6.7 Bases for the Hydrogen Recombiners were revised to describe the ANO-1 installed configuration. The ANO-1 design features two, independent, 100% capacity hydrogen recombiners installed inside the reactor building. The design also is based on a 4 v/o (volume percent) minimum flammability concentration for hydrogen which is more conservative than 4.1 v/o. These changes are consistent with current license basis.

18. NUREG 3.6.3 - Incorporated TSTF-269, Rev 2.

19. NUREG SR 3.6.3.4 - The Frequency of ITS SR 3.6.3.3 is modified to clarify that the position of the reactor building (RB) isolation valves inside the RB needs to be verified only once within any 92 day period when entering MODE 4 from MODE 5. Transition to MODE 4 from MODE 5 occurs when returning to power from an outage. This end of outage timeframe has strong administrative controls on component status such as valve position. Therefore, validating valve position is needed only once during this time since these administrative controls, coupled with controls on RB entry, provide assurance that valve position will remain as validated once startup from an outage is initiated. This application of "once" to a Frequency is consistent with NUREG-1430 as discussed in NUREG Example 1.4-2. The Bases discussion is modified as required.

20. NUREG SR 3.6.4.1 - Surveillance of the reactor building pressure is not a requirement in the CTS. NUREG SR 3.6.4.1 will be adopted as ITS SR 3.6.4.1 with an extension of the Frequency from 12 hours to 24 hours. The 24 hour Frequency takes into account ANO-1 operating experience related to trending of reactor building pressure variations during the applicable ITS MODES. Further, this Frequency is consistent with the Frequency for NUREG SR 3.6.5.5 for reactor building temperature. The appropriate Bases were updated.

21. NUREG 3.6.6 is renumbered to ITS 3.6.5 because NUREG 3.6.5, Containment Temperature, was not adopted [Ref. DOD 12]. ITS 3.6.5 is modified by the addition of a Note and revision of the Conditions and Required Actions. The NUREG is modified to retain the CTS 3.3.1 and 3.3.4 requirements for two trains of reactor building (RB) spray and two trains of RB cooling OPERABLE during MODES 1 and 2, and one train of RB spray and one train of RB cooling OPERABLE during MODES 3 and 4. During MODES 3 and 4, the potential energy release of the high energy systems inside the reactor building is significantly less than during MODES 1 and 2. Requiring the availability of one train of RB spray and one train of RB cooling during MODES 3 and 4 is considered adequate given the reasons above and the limited time spent in these MODES. These changes are consistent with current license basis.

    ITS SR 3.6.5.1, SR 3.6.5.4, SR 3.6.5.5, SR 3.6.5.6, and SR 3.6.5.8 were modified to incorporate the requirements for required trains of reactor building spray during MODES 3 and 4. These changes are consistent with current license basis.

The NUREG 3.6.7 Applicability is modified to MODES 1 and 2 for ITS 3.6.6 in order to retain the CTS 3.3.4 requirements for availability of sodium hydroxide. NUREG 3.6.7 Condition B is modified to retain the CTS requirements and to make ITS 3.6.6 Condition B compatible with ITS 3.6.5 Condition D. These changes are consistent with current license basis.

The NUREG 3.6.7 Bases and LCO 3.6.6 Bases are both marked up to reflect these changes. These changes are consistent with current license basis.

22. NUREG 3.6.1 Bases – The Applicable Safety Analyses (ASA) discussion includes a statement that satisfactory leakage rate test results are a requirement of containment OPERABILITY. This statement is also paraphrased and included, more appropriately, in the discussion for the NUREG 3.6.1 LCO. The ITS will delete the redundant sentence from the ASA and retain the appropriate statement in the LCO discussion. This change constitutes an editorial preference.

23. A maximum volume for the sodium hydroxide tank was deleted from NUREG SR 3.6.7.2 (renumbered in the ITS as SR 3.6.6.2). In CTS 3.3.4(B) only the minimum NaOH tank volume is required. The ANO-1 calculation assumption, upon which a maximum tank volume would be based, exceeds the physical volume of the NaOH tank and is therefore moot. This change complies with the CTS.

24. NUREG 3.6.2 - The NUREG 3.6.2, Condition A, Required Action Note 2 is not adopted in the ITS. This NUREG Note addresses requirements for airlock use that are considered in conflict with the allowance of ITS 3.6.2 ACTIONS Note 1 (which allows unlimited entry and exit to perform maintenance on an affected airlock). Furthermore, NUREG-1430 3.6.2, Condition A, Required Action Note 2 is inconsistent with the requirements inferred by the CTS. Administrative controls have to date proven adequate to control entry and exit requirements during airlock maintenance. It is proposed to retain these controls under licensee administrative control. This change is consistent with current license basis.

25. NUREG 3.6.6 Bases - The ITS 3.6.5 Bases for ACTIONS A.1 and B.1 are revised to emphasize that the remaining OPERABLE reactor building (RB) spray train is redundant for reactor building cooling and for support of the Spray Additive System function of iodine removal. In the Conditions associated with these ACTIONS, the OPERABLE RB cooling trains are also redundant for cooling but not iodine removal. This change is consistent with current license basis.

26. NUREG 3.6.6 Bases - The ITS SR 3.6.5.2 Bases are revised to reflect the current license basis of the unit. The installed configuration of the reactor building (RB) cooling trains is not able to indicate the status of individual train components for vibration or blockage, etc. This change is consistent with current license basis.

27. NUREG 3.6.6 Bases - The ITS 3.6.5 Bases Background discussion was revised to identify the version of the General Design Criteria used during the design ANO Unit 1. The bulk of ANO-1 design, including the general design criteria, occurred prior to the 1971 inception of 10 CFR 50, Appendix A, and the titles and criteria statements in SAR Section 1.4 are slightly different. This change is consistent with current license basis.

28. NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

    The 10 CFR 50.36 Criterion satisfied by the ITS LCOs was modified to preserve consistency with the ANO-1 license basis. The NUREG Criterion specified were modified to be consistent with the analysis assumptions regarding equipment availability and operating condition (i.e., MODE).

29. Not used.

30. NUREG 3.6.3 Bases - The NUREG 3.6.3 Bases Background, Applicable Safety Analysis, and SR 3.6.3.5 Bases discussions are revised for the ITS to address that the ANO-1 analyses for reactor building isolation set no specific values for reactor building isolation valve response times. Consideration of the response of the reactor building isolation valves is satisfactorily addressed by assuring valve response times are maintained in accordance with the industry standards for sizing valve operators. This is consistent with the current ANO-1 licensing basis and supports the objective to minimize the potential leakage paths to the environment.

31. NUREG 3.6.6 Bases - The ITS SR 3.6.5.3 Bases discussion is revised to emphasize the relationship of the 1200 gpm service water flow to the original design intent for the reactor building coolers. This change is consistent with current license basis.

32. NUREG 3.6.2 Bases - The ITS 3.6.2 Background discussion deletes this sentence since it discusses airlock operation in MODES 5 and 6. Guidance is provided in ITS 3.9, "Refueling Operations."

33. NUREG 3.6.6 Bases - The reference to "relatively cold" borated water is deleted from the ITS 3.6.5 Bases Background discussion. A subjective value for the fluid temperature is not pertinent to this Bases discussion.

34. NUREG 3.6 Bases - Bases discussions are revised to detail attributes of the ANO-1 design bases. The ANO-1 Design Basis Accidents consider that Power Operation (ITS MODE 1) conditions are the initial conditions.

35. NUREG 3.6.3 ACTIONS Note 1 has an inclusive intent (penetration flow paths) and an exclusive intent (purge valve flow paths). In the CTS, the option to open penetration flow paths under administrative control is provided by the "as required" allowance in definition 1.7 item c. However, the exclusion of the purge valve path is provided by CTS 3.23.1 which states: "shall be closed ... whenever containment integrity is required ...".

36. NUREG SR 3.6.1.2 and its associated Bases are not incorporated in the proposed ITS. Amendment 199, September 9, 1999, that revised the current reactor building structural integrity requirements did not include a similar SR since reactor building structural integrity is verified in accordance with Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code, incorporated by reference in 10CFR50.55a. Additional statements have been added to the LCO discussion in the Bases of ITS 3.6.1 to provide assurance that structural integrity is also a measure of reactor building operability. This change is consistent with the current license basis.

37. NUREG 3.6.4 Background and NUREG 3.6.6 ASA are modified to remove discussions of inadvertent containment spray issues. The functional and design requirements related to structural integrity during an inadvertent containment spray are not related to the 10 CFR 50.36 criteria for Technical Specification process parameters. The values retained (and discussed in the Bases) relate the DBA and ECCS analyses.

38. NUREG 3.6.3 Required Action A.1 Completion Time (and its Bases) is revised. The time allowed to isolate a penetration with one inoperable isolation valve and one operable valve is revised to 48 hours to reflect the current licensing basis in the CTS.

39. NUREG 3.6.4 statement in the Bases has been provided to explicitly establish that the safety analysis parameters presented in the ITS do not contain allowances for instrumentation error. This change is considered to be administrative in nature.

40. NUREG 3.6.3 LCO Bases discussing normally closed isolation valves is revised to correct erroneous descriptions. The revised discussion mirrors the requirements found in the related surveillances – SR 3.6.3.2 and SR 3.6.3.3.

CTS

Reactor Building
~~Containment~~
3.6.1 —(1)

REACTOR BUILDING
3.6 ~~CONTAINMENT~~ SYSTEMS

Reactor Building
3.6.1 ~~Containment~~

The reactor building
LCO 3.6.1 ~~Containment~~ shall be OPERABLE. L.7a&e 3.6.1

APPLICABILITY: MODES 1, 2, 3, and 4. 3.6.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. ~~Containment~~ Reactor building inoperable.	A.1 Restore ~~containment~~ reactor building to OPERABLE status.	1 hour	3.6.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours	3.6.1 3.6.1

CTS

Reactor Building

~~Containment~~

(1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations reactor building and leakage rate testing except for ~~containment~~ air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~The leakage rate acceptance criterion is ≤ 1.0 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 L_a for the Type B and Type C tests, and < 0.75 L_a for the Type A test.~~	~~-----NOTE----- SR 3.0.2 is not applicable~~ In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ The Reactor Building Leakage Rate Testing Program.
~~SR 3.6.1.2~~	~~Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.~~	~~In accordance with the Containment Tendon Surveillance Program~~

4.4.1.1
4.4.1.1A
4.4.1.2
4.4.1.2.5
1.7e

(2)

(36)

CTS
1

REACTOR BUILDING
3.6 ~~CONTAINMENT~~ SYSTEMS

Reactor Building
3.6.2 ~~Containment~~ Air Locks 1.7a & b

reactor building
LCO 3.6.2 ~~[Two]~~ ~~containment~~ air lock~~[s]~~ shall be OPERABLE. 3.6.1

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Entry and exit is permissible to perform repairs on the affected air lock components. NA
2. Separate Condition entry is allowed for each air lock. NA
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building ~~Containment~~," when air lock leakage results in exceeding the overall ~~containment~~ reactor building leakage rate acceptance criteria. NA

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more reactor building ~~containment~~ air locks with one ~~containment~~ air lock door inoperable.	-----NOTE~~S~~----- ~~1.~~ Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. ~~2. Entry and exit is permissible for 7 days under administrative controls [if both air locks are inoperable].~~ -----	NA 24
	A.1 Verify the OPERABLE door is closed in the affected air lock. AND	1 hour 3.6.1

(continued)

CTS

H 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. (continued)	A.2 Lock the OPERABLE door closed in the affected air lock. AND	24 hours	NA
	A.3 ---------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --------------------- Verify the OPERABLE door is locked closed in the affected air lock.	 Once per 31 days	NA NA
B. One or more reactor building ~~containment~~ air locks with ~~containment~~ reactor building air lock interlock mechanism inoperable.	-------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of ~~containment~~ the reactor building is permissible under the control of a dedicated individual. ---------------------------- B.1 Verify an OPERABLE door is closed in the affected air lock. AND	 1 hour (continued)	NA

CTS

Reactor Building

~~Containment~~ Air Locks
3.6.2

H 1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
B. (continued)	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours	
	AND			
	B.3	---------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. -------------------- Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days	
C. One or more reactor building ~~containment~~ air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall reactor building ~~containment~~ leakage rate per LCO 3.6.1.	Immediately	NA
	AND			
	C.2	Verify a door is closed in the affected air lock.	1 hour	
	AND			
	C.3	Restore air lock to OPERABLE status.	24 hours	
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours	3.6.1
	AND			
	D.2	Be in MODE 5.	36 hours	3.6.1

CTS

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.6.2.1	-------NOTES------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against applicable to acceptance criteria ~~of~~ SR 3.6.1.1 ~~in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions~~. ---------------		NA NA
	Perform required air lock leakage rate testing in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~The acceptance criteria for air lock testing are:~~ ~~a. Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~ ~~b. For each door, leakage rate is ≤ [.01 L_a] when tested at ≥ [10.0 psig].~~ the Reactor Building Leakage Rate Testing Program	~~-----NOTE----- SR 3.0.2 is not applicable ----------~~ In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~	4.4.1.2 4.4.1.2.5 2
SR 3.6.2.2	~~-------NOTE------- Only required to be performed upon entry or exit through the containment air lock. ----------~~ Verify only one door in the air lock can be opened at a time.	18 months ~~184 days~~	8 NA

CTS

 ①

3.6 REACTOR BUILDING ~~CONTAINMENT~~ SYSTEMS

3.6.3 Reactor Building ~~Containment~~ Isolation Valves — 1.7c & d 3.23.1

LCO 3.6.3 Each reactor building ~~containment~~ isolation valve shall be OPERABLE. — 3.6.1 3.6.6 3.23.1

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

⑦ ㉟ 1.7.C. 3.23.1

-----NOTES-----

1. Penetration flow paths~~,~~ except for ~~48 inch~~ purge valve penetration flow paths~~,~~ may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path. — NA
3. Enter applicable Conditions and Required Actions for system(s) made inoperable by reactor building ~~containment~~ isolation valves. — NA
4. ~~Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.~~ ⑦

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE--------- Only applicable to penetration flow paths with two reactor building ~~containment~~ isolation valves. --------------------- One or more penetration flow paths with one reactor building ~~containment~~ isolation valve inoperable ~~(except for purge valve leakage not within limit)~~.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	~~4~~ 48 hours ㊳

NA 3.6.6

⑦

(continued)

CTS

1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued) <INSERT 3.6-8A>	A.2 ----------NOTE---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. ------------------------- Verify the affected penetration flow path is isolated.	18 Once per 31 days for isolation devices outside ~~containment~~ the reactor building AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside ~~containment~~ the reactor building
B. ---------NOTE--------- Only applicable to penetration flow paths with two ~~containment~~ reactor building isolation valves. ---------------------- One or more penetration flow paths with two ~~containment~~ reactor building isolation valves inoperable ~~(except for purge valve leakage not within limit)~~.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

NA

N/A

NA

NA

NA

7

(continued)

<INSERT 3.6-8A>

2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

CTS

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Only applicable to penetration flow paths with only one reactor building ~~containment~~ isolation valve and a closed system. --------------------- One or more penetration flow paths with one ~~containment~~ reactor building isolation valve inoperable. <INSERT 3.6-9A>	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND C.2 ---------NOTE--------- 5 1 Isolation devices in high radiation areas may be verified by use of administrative means. ------------------ Verify the affected penetration flow path is isolated.	~~[4]~~ 72 hours Once per 31 days
~~[D. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.~~	~~D.1 Isolate the affected penetration flow path by use of at least one [closed and de-activated automatic valve, closed manual valve, or blind flange]. AND~~	~~24 hours]~~

NA 6

18

7

(continued)

<INSERT 3.6-9A>

2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

CTS

1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	--------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. ------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	AND D.3	Perform SR 3.6.3.6 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per [ ] days
D ~~E~~. Required Action and associated Completion Time not met.	D ~~E~~.1	Be in MODE 3.	6 hours
	AND D ~~E~~.2	Be in MODE 5.	36 hours

7

3.6.1

7

3.6.1
3.6.6

 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.6.3.1	Verify each reactor building ~~[48] inch~~ purge isolation valve is ~~sealed~~ closed ~~except for one purge valve in a penetration flow path while in Condition D of the LCO~~.	31 days	4.26.1 7
~~SR 3.6.3.2~~	~~Verify each [8] inch purge valve is closed except when the [8] inch purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.~~	~~31 days~~	7
SR 3.6.3.~~3~~ 2	-------------------NOTE------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each ~~containment~~ the reactor building isolation manual valve and blind flange that is located outside ~~containment~~ reactor building, and not locked, Sealed, or otherwise Secured, and is required to be closed during accident conditions is closed, except for ~~containment~~ isolation valves that are open under administrative controls.	31 days	NA 3.6.5 4

(continued)

CTS

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.6.3.~~1~~ 3	---NOTE--- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ------ Verify each ~~containment~~ reactor building isolation manual valve and blind flange that is located inside ~~containment~~ the reactor building and required to be closed during accident conditions is closed, except for ~~containment~~ reactor building isolation valves that are open and not locked, sealed, or otherwise secured, under administrative controls.	Once Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days	NA L9 3.6.5 4
SR 3.6.3.~~5~~ 4	Verify the isolation time of ~~each power operated and~~ each automatic power operated ~~containment~~ reactor building isolation valve is within limits.	In accordance with the Inservice Testing Program ~~or 92 days~~	5 4.4.1.4
~~SR 3.6.3.6~~	~~Perform leakage rate testing for containment purge valves with resilient seals.~~	~~184 days AND Within 92 days after opening the valve~~	7
SR 3.6.3.~~7~~ 5	Verify each automatic ~~containment~~ reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	~~[18]~~ months	Tbl 4.1-2 #8

(continued)

CTS

Reactor Building

~~Containment~~ Isolation Valves
3.6.3

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[SR 3.6.3.8	Verify each [ ] inch containment purge valve is blocked to restrict the valve from opening > [50]%.	[18] months]

7

CTS

Reactor Building 1

~~Containment~~ Pressure
3.6.4

REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS

Reactor Building

3.6.4 ~~Containment~~ Pressure

Reactor Building -1.0

LCO 3.6.4 ~~Containment~~ pressure shall be ≥ ~~[-2.0]~~ psig and ≤ ~~[~~+3.0~~]~~ psig. 3.6.4

APPLICABILITY: MODES 1, 2, 3, and 4. 3.6.4

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. Reactor building ~~Containment~~ pressure not within limits.	A.1	Restore Reactor building ~~containment~~ pressure to within limits.	1 hour	3.6.4
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours	3.6.4
	AND			
	B.2	Be in MODE 5.	36 hours	3.6.4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.4.1	Verify Reactor building ~~containment~~ pressure is ~~within limits.~~ ≥ -1.0 psig and ≤ +3.0 psig.	24 ~~12~~ hours	20 NA edit

CTS

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ [130]°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment average air temperature not within limit.	A.1	Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

12

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	24 hours

CTS

1

REACTOR BUILDING
3.6 ~~CONTAINMENT~~ SYSTEMS

Reactor Building
3.6.~~6~~ 5 ~~Containment~~ Spray and Cooling Systems

LCO 3.6.~~6~~ 5 Two reactor building ~~containment~~ spray trains and two reactor building ~~containment~~ cooling trains shall be OPERABLE. 3.3.1 (A&B) 3.3.1 (I) 3.3.4 3.3.4 (D) 21

<Insert 3.6-16A>

APPLICABILITY: MODES 1, 2, 3, and 4. 3.3.1 3.3.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~containment~~ reactor building spray train inoperable in MODE 1 or 2.	A.1 Restore reactor building ~~containment~~ spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO	3.3.6 3.3.7(E) NA
D ~~B~~. Required Action and associated Completion Time of Condition A, B, or C not met.	D ~~B~~.1 Be in MODE 3. ~~AND~~ ~~B.2 Be in MODE 5.~~	6 hours ~~84 hours~~	3.3.6 3.3.7(C) (D) & (E) 21
B ~~C~~. One ~~[required]~~ reactor building ~~containment~~ cooling train inoperable in MODE 1 or 2.	B ~~C~~.1 Restore ~~[required]~~ reactor building ~~containment~~ cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO	3.3.7(C) 3.3.7(D) 3.3.7(E) NA

(continued)

<INSERT 3.6-16A> **CTS**

--------------------------NOTE--------------------------------

Only one train of reactor building spray and one train of reactor building cooling are required to be OPERABLE during MODES 3 and 4. 3.3.1(A) 3.3.1(B)

ITS 3.6.5 ACTIONS NUREG Revisions Reviewer clarification sheet

A.	One reactor building spray train inoperable in MODE 1 or 2.	A.1	Restore reactor building spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
B.	One reactor building cooling train inoperable in MODE 1 or 2.	B.1	Restore reactor building cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
C.	Two reactor building cooling trains inoperable in MODE 1 or 2.	C.1	Restore one reactor building cooling train to OPERABLE status.	72 hours
D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
E.	One required reactor building spray train inoperable in MODE 3 or 4. OR One required reactor building cooling train inoperable in MODE 3 or 4.	E.1	Restore required inoperable train to OPERABLE status.	36 hours
F.	Required Action and associated Completion Time of Condition E not met.	F.1	Be in MODE 5.	36 hours
G.	Two reactor building spray trains inoperable in MODE 1 or 2. OR Any combination of three or more trains inoperable in MODE 1 or 2. OR One required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4.	G.1	Enter LCO 3.0.3.	Immediately

Insert after NUREG pg. 3.6-16

CTS

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
C ~~D~~. Two ~~[required]~~ reactor building ~~containment~~ cooling trains inoperable in MODE 1 or 2.	C ~~D~~.1 Restore one ~~[required]~~ reactor building ~~containment~~ cooling train to OPERABLE status.	72 hours	3.3.7(D)
<Insert 3.6-17A>			
F ~~E~~. Required Action and associated Completion Time of Condition ~~C~~ ~~or D~~ E not met.	~~E.1 Be in MODE 3.~~ ~~AND~~	~~6 hours~~	21
	~~E.2~~ F.1 Be in MODE 5.	36 hours	3.3.6
G ~~F~~. Two reactor building ~~containment~~ spray trains inoperable in MODE 1 or 2. OR Any combination of three or more trains inoperable in MODE 1 or 2.	G ~~F~~.1 Enter LCO 3.0.3.	Immediately	NA
<INSERT 3.6-17B>			

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.~~6~~ 5.1	Verify each reactor building ~~containment~~ spray manual, power operated, and automatic valve in ~~the~~ each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days	NA 21

(continued)

<INSERT 3.6-17A>

CTS

E. One required reactor building spray train inoperable in MODE 3 or 4. OR One required reactor building cooling train inoperable in MODE 3 or 4.	E.1 Restore required inoperable train to OPERABLE status.	36 hours	3.3.6

<INSERT 3.6-17B>

OR One required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4.		

CTS

Reactor Building

~~Containment~~ Spray and Cooling Systems
3.6.~~6~~ 5

⊢ 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	CTS
SR 3.6.~~6~~ 5.2	Operate each ~~[required]~~ reactor building ~~containment~~ cooling train fan unit for ≥ 15 minutes.	31 days	4.5.2.1.2 (b)
SR 3.6.~~6~~ 5.3	Verify each ~~[required]~~ reactor building ~~containment~~ cooling train cooling water flow rate is ≥ ~~[1780]~~ 1200 gpm.	31 days	4.5.2.1.2 (a) 4.5.2.1.2(a)(1)
SR 3.6.~~6~~ 5.4	Verify each required reactor building ~~containment~~ spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	4.5.2.2.1 ⊢ 21
SR 3.6.~~6~~ 5.5	Verify each automatic reactor building ~~containment~~ spray valve in each required ~~the~~ flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[18]~~ 18 months	4.5.2.1.1(a) ⊢ 21
SR 3.6.~~6~~ 5.6	Verify each required reactor building ~~containment~~ spray pump starts automatically on an actual or simulated actuation signal.	~~[18]~~ 18 months	4.5.2.1.1 (a) ⊢ 21
SR 3.6.~~6~~ 5.7	Verify each ~~[required]~~ reactor building ~~containment~~ cooling train starts automatically on an actual or simulated actuation signal.	~~[18]~~ 18 months	4.5.2.1.2 (c) & (c)(1) & (c)(3)

(continued)

CTS

Reactor Building
~~Containment~~ Spray and Cooling Systems
3.6.5

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.8	Verify each required train spray nozzle is unobstructed.	~~At first refueling~~ ~~AND~~ 10 years

21 4.5.2.11 (b)

CTS

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REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS

3.6.~~7~~ 6 Spray Additive System

LCO 3.6.~~7~~ 6 The Spray Additive System shall be OPERABLE. 3.3.4 & (C) & (D)

APPLICABILITY: MODES 1, and 2~~, 3, and 4~~. 3.3.4 21

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. ~~AND~~ ~~B.2 Be in MODE 5.~~	6 hours ~~84 hours~~

3.3.6

3.3.6

21

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.~~7~~ 6.1	Verify each ~~spray additive~~ System manual, power operated, ~~and~~ automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

9 NA

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.6.~~7~~ 6.2	Verify ~~spray additive~~ Sodium hydroxide tank solution volume is ≥ ~~[12,970]~~ 9000 gal ~~and ≤ [13,920] gal~~.	184 days	9 23 3.3.4 (B)
SR 3.6.~~7~~ 6.3	Verify ~~spray additive~~ Sodium hydroxide tank ~~[NaOH]~~ solution concentration is ~~≥ [60,000 ppm] and ≤ [65,000 ppm]~~. > 5.0 wt% and < 16.5 wt% NaOH	184 days	3.3.4(B) T 4.1-3 #6.
SR 3.6.~~7~~ 6.4	Verify each ~~spray additive~~ System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	~~[18]~~ 18 months	9 NA
~~SR 3.6.7.5~~	~~Verify Spray Additive System flow [rate] from each solution's flow path.~~	~~5 years~~	13

CTS

 1 edit

REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS

3.6.~~8~~ 7 Hydrogen Recombiners ~~(if permanently installed)~~ 3.14.1

LCO 3.6.~~8~~ 7 Two hydrogen recombiners shall be OPERABLE. 3.14.1

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One hydrogen recombiner inoperable.	A.1 --------NOTE--------- LCO 3.0.4 is not applicable. --------------------- Restore hydrogen recombiner to OPERABLE status.	30 days	NA 3.14.2
~~B. Two hydrogen recombiners inoperable.~~	~~B.1 Verify by administrative means that the hydrogen control function is maintained.~~ ~~AND~~ ~~B.2 Restore one hydrogen recombiner to OPERABLE status.~~	~~1 hour AND Every 12 hours thereafter~~ ~~7 days~~	10
~~C~~ B. Required Action and associated Completion Time not met.	~~C~~ B.1 Be in MODE 3.	6 hours	3.14.2

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.8.1 7	Perform a system functional test for each hydrogen recombiner.	18 [24] months	4.12.1 a.
SR 3.6.8.2 7	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	18 [24] months	4.12.1 b.
SR 3.6.8.3 7	Perform a resistance to ground test for each heater phase.	18 [24] months	4.12.1 b.

1

REACTOR BUILDING

B 3.6 CONTAINMENT SYSTEMS

Reactor Building

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building (RB), its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

of
liner and
Structure
reactor building
the reactor building
radioactivity
an
edit
edit
edit

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. For containments with ungrounded tendons, the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

where
The reactor building design includes
edit
edit

reinforced
structure

The concrete RB is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

, Option B
2

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

1. capable of being closed by an OPERABLE automatic containment isolation system, or

edit

Insert B3.6-1A

(continued)

<INSERT B3.6-1A>

except as provided in LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," LCO 3.3.6, "ESAS Manual Initiation," and LCO 3.3.7, "ESAS Actuation Logic"

Reactor Building

~~Containment~~

1

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves ~~secured~~ in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves"; edit

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; edit

c. the ~~All~~ equipment ~~hatches~~ is ~~are~~ closed; ~~and~~ edit.

~~[d. The pressurized sealing mechanism associated with each penetration, except as provided in LCO 3.6.[ ], is OPERABLE.]~~

APPLICABLE SAFETY ANALYSES

edit

The ~~safety~~ design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

edit

The DBAs that result in a challenge to containment ~~OPERABILITY~~ from high pressures and temperatures are a loss of coolant accident (LOCA), and a steam line break ~~[and a rod ejection accident (REA)]~~ (Ref. 2). In addition, release of significant fission product~~s~~ ~~radioactivity~~ within containment can occur from a LOCA ~~or REA~~. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for ~~the~~ ~~DBAs involving release of~~ a LOCA fission product ~~radioactivity~~, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[0.25]~~ 0.2% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable leakage rate at the calculated maximum peak containment pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be ~~[0.25]~~ 0.2% per day in the safety analysis at P_a = ~~[53.9]~~ 54.0 psig (Ref~~s~~. 3). 2 22

~~Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.~~

In MODES 1 and 2, ~~The~~ the containment satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building satisfies Criterion 4 of 10 CFR 50.36. 28

(continued)

1

BASES (continued)

LCO

<INSERT B 3.6-3A>

Containment OPERABILITY is maintained by limiting leakage to ~~≤ 1.0 L_a, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the combined Type B and C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a.~~ Compliance ~~with this LCO~~ will ensure a containment configuration, ~~including equipment hatches,~~ that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

In conjunction with LCO 3.6.2

2 36 edit

Individual leakage rates specified for the containment air lock (LCO 3.6.2) ~~[and purge valves with resilient seals (LCO 3.6.3)]~~ are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of ~~Appendix J~~ 1.0 La.

edit

2

APPLICABILITY

<Insert B3.6-3B>

~~In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment.~~ In MODES 5 and 6, the probability and consequences of ~~these~~ events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "~~Containment~~ Reactor Building Penetrations."

34

edit

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining ~~containment~~ reactor building OPERABILITY during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

edit

(continued)

<INSERT B3.6-3A>

... within limits specified by the Reactor Building Leakage Rate Testing Program. Reactor building OPERABILITY for leakage is attained by ensuring that the equipment hatch and both doors of the personnel and emergency air locks are closed and sealed, except as appropriate for maintenance activities, and that the other isolation devices are closed, deactivated in the closed position, or OPERABLE as required. Reactor building OPERABILITY is also maintained by monitoring the deviation of key design parameters of the RB structure from the original design configuration and ensuring that structural limits are not exceeded. The structural monitoring programs are performed in accordance with Subsection IWL of ASME Section XI Boiler and Pressure Vessel Code, as referenced by 10CFR50.55a, and 10CFR50, Appendix J (Ref. 1).

<INSERT B3.6-3B>

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Actions and associated Completion Times are not met

~~If containment cannot be restored to OPERABLE status within the required Completion Time,~~ the ~~plant~~ Unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ Unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~plant~~ conditions from full power conditions in an orderly manner and without challenging ~~plant~~ systems. edit edit edit edit

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.~~ (2) ⟨Insert B3.6-4 A⟩ Failure to meet air lock ~~and purge valve with resilient seal~~ leakage limits specified in LCO 3.6.2 ~~and LCO 3.6.3~~ does not invalidate the acceptability of these overall leakage determinations unless ~~their~~ its contribution ~~to~~ overall Type A, B, and C leakage causes ~~that~~ to exceed limits. (7) edit As left leakage prior to the first startup after performing a required ~~10 CFR 50, Appendix J,~~ leakage test is required to be ≤ 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. (2) ⟨Insert B3.6-4 B⟩ At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. ⟨Insert B3.6-4C⟩ At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by ~~Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~ (2) ⟨Insert B3.6-4D⟩ These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

~~SR 3.6.1.2~~

~~For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and~~ (36)

~~(continued)~~

<Insert B3.6-4 A>

the Reactor Building Leakage Rate Testing Program.

<Insert B3.6-4 B>

following an outage or shutdown that included Type B or C testing

<Insert B3.6-4 C>

following an outage or shutdown that included Type A testing

<Insert B3.6-4 D>

the Reactor Building Leakage Rate Testing Program, which implements 10CFR50, App. J, and includes more frequent testing of the reactor building purge isolation valves.

1

BASES

~~SURVEILLANCE REQUIREMENTS~~	~~SR 3.6.1.2 (continued)~~ ~~Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4).~~

36

REFERENCES

1. 10 CFR 50, Appendix J.
2. ~~F~~SAR, Chapter 14 ~~Sections [14.1 and 14.2]~~. edit
3. ~~F~~SAR, Section ~~[5.6]~~ 5.2. edit
4. 10CFR 50.36. ~~Regulatory Guide 1.35, Revision [1].~~ 36 28

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

, also known as the personnel air lock and the emergency (or escape) air lock

11

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

24

Each air lock is nominally a right circular cylinder, ~~10 ft in diameter~~, with a door at each end. The doors are interlocked to prevent simultaneous opening. ~~During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary.~~ Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

32

~~Each personnel air lock door is provided with limit switches that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.~~

3

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness ~~is~~ are essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

edit

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), and a steam line break~~, and a rod ejection accident~~ (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[0.25]%~~ ≤ 0.2 % of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) following a DBA LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. ~~L_a is [0.25]% per day and P_a is [53.9] psig resulting from the limiting design basis LOCA.~~ edit 2 2

In MODES 1 and 2, the containment air locks satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building air locks satisfy Criterion 4 of 10CFR50.36. 28

LCO

Each containment air lock forms part of the containment pressure boundary. As a part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE (ie., Closed and Sealed). The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure and Sealing of a single door in each air lock ~~is~~ sufficient ~~to~~ provides ~~a leak tight~~ leakage barrier following postulated events. Nevertheless, both doors are ~~kept~~ normally closed and Sealed when the air lock is not being used for normal entry into ~~and~~ or exit from containment. edit edit edit

H 1

BASES (continued)

APPLICABILITY

<Insert B3.6-8A>

~~In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment.~~ In MODES 5 and 6, the probability and consequences of ~~these~~ events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "~~Containment~~ Reactor Building Penetrations."

H 34

edit

ACTIONS

The ACTIONS are modified by a but capable of being swung Note that allows entry and exit to perform repairs on the affected air lock component. and the airlock barrel If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that ~~the air lock~~ an inoperable inner door be accessed from inside ~~primary~~ containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). ~~The ability to open~~ Opening the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ~~ALARA~~ conditions permit, entry and exit should be via an OPERABLE air lock.

edit

edit

edit

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

(continued)

<INSERT B3.6-8A>

In MODES 1, 2, 3 and 4, the reactor building air lock OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES.

1

BASES

ACTIONS (continued)

A.1, A.2, and A.3

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock.

This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the remaining OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small. edit

The Required Actions have been modified by a ~~two~~ Note~~s~~. that ~~Note 1~~ clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of this ~~Note 1~~ does not affect tracking the Completion Time from the 24

(continued)

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BASES

ACTIONS

A.1, A.2, and A.3 (continued)

initial entry into Condition A; only the requirement to comply with the Required Actions. ~~Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.~~ 24

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from the containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the

(continued)

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BASES

ACTIONS

B.1, B.2, and B.3 (continued)

probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a ~~plant~~ unit shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits. edit

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock~~(s)~~ must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock. edit

D.1 and D.2

If the Required Actions and associated Completion Times are not met

~~If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time,~~ the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, edit edit edit

(continued)

Reactor Building 1

BASES

ACTIONS

D.1 and D.2 (continued)

unit

edit

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

unit

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

the Reactor Building Leakage Rate Testing Program

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. 2 2

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the overall containment leakage rate. 2

Combined Type B and C

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

not normally

will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is ~~only~~ challenged when the containment air lock door is ~~opened~~, this test is only required to be performed ~~upon entering or exiting a containment air lock but is not required more frequently than every 184 days~~. The ~~184 day~~ 18 month Frequency is based on engineering judgment and is considered adequate ~~in view of other indications of door and interlock mechanism status available to operations personnel~~.

Insert B3.6-13 A

Insert B3.6-13 B

Insert B3.6-13 C

8

8

2

REFERENCES

1. 10 CFR 50, Appendix J, Option B
2. ~~F~~SAR, ~~Sections [14.1 and 14.2]~~ Chapter 14 — edit
3. ~~F~~SAR, ~~Section [6.6]~~ Chapter 5 — edit
4. 10 CFR 50.36. — 28

<Insert B3.6-13A>

used for entry and exit (procedures require strict adherence to single door opening),

<Insert B3.6-13B>

every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, and the potential for loss of reactor building OPERABILITY if the Surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience.

<Insert B3.6-13C>

given that the interlock is not expected to be challenged during use of the airlock.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position~~, including~~ check valves ~~with flow through the valve secured~~, blind flanges, and closed systems are considered passive devices. ~~Check valves, or other~~ automatic valves designed to close following an accident without operator action, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

edit

Containment isolation occurs upon receipt of a high containment pressure ~~or diverse containment isolation~~ signal. The containment isolation signal closes automatic containment isolation valves in fluid penetrations not required for operation of engineered safeguard systems to prevent leakage of radioactive material. ~~Upon actuation of high pressure injection,~~ automatic containment valves ~~also~~ isolate systems ~~not required for containment or Reactor Coolant System (RCS) heat removal~~. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following a Design Basis Accident (DBA).

Also, upon receipt of a low RCS pressure signal certain

edit

edit

edit

OPERABILITY of the containment isolation valves (and blind flanges) supports containment OPERABILITY during accident conditions.

(continued)

 1

BASES

BACKGROUND (continued)

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated ~~within the time limits assumed in the safety analysis.~~ Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analysis will be maintained. 30

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Purge System was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into containment. The Containment Purge System consists of one [48] inch line for exhaust and one [48] inch line for supply, with supply and exhaust fans~~, capable of purging the containment atmosphere at a rate of approximately [50,000] ft³/min. This flow rate is sufficient to reduce the airborne radioactivity level within containment to levels defined in 10 CFR 20 (Ref. 1) for a 40 hour workweek within 2 hours of purge initiation during reactor operation.~~ The containment purge supply and exhaust lines each contain two isolation valves that receive ~~an isolation signal on~~ a ~~unit vent high radiation condition~~. 7

24

reactor building isolation Signal

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large containment leakage path introduced by these [48] inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the containment design leakage rate of [0.25]% of containment air weight per day (L_a) (Ref. 2). ~~Because of their large size,~~ the [48] inch purge valves ~~in some units~~ are not ~~qualified~~ for automatic closure from their open position under DBA conditions. Therefore, the [48] inch purge valves are maintained ~~sealed~~ closed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. 7

24

≤0.2

24

1

tested

24

with the hand switch keys removed

Supply and exhaust

~~The [8 inch] containment minipurge valves operate to:~~

~~a. Reduce the concentration of noble gases within containment prior to and during personnel access; and~~

~~b. Equalize internal and external pressures.~~

~~Since the minipurge valves are designed to meet the requirements for automatic containment isolation valves,~~ 7

(continued)

 1

BASES

BACKGROUND (continued)

~~these valves may be opened as needed in MODES 1, 2, 3, and 4.~~ 7

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports isolation ~~leak tightness~~ of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO. edit

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), and a main steam line break ~~and a rod ejection accident~~ (Ref. 3). 2 edit In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close ~~within the required isolation time following event initiation~~. 30 This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the ~~[48]~~ 24 inch purge valves are closed at event initiation.

Insert B3.6-16A

~~The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.~~ 30

~~The single-failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the containment purge valves. Two valves in a series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.~~ 7

(continued)

<Insert B3.6-16A>

The LOCA analysis assumes a fixed amount of core inventory escapes. No mechanistic scenario is evaluated to determine what portion of the inventory is released prior to closure of the reactor building isolation valves. Industry standards for sizing valve operators govern the closure times of the reactor building isolation valves.

 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single-failure criterion remains applicable to the containment purge valves because of failure in the control circuit associated with each valve. Again, the purge system valve design prevents a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.~~ 7

In MODES 3 and 4, the reactor building isolation valves satisfy Criterion 4 of 10 CFR 50.36.

In MODES 1 and 2, The ~~containment~~ isolation valves satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 28

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary ~~during a DBA~~. 34

edit

The automatic ~~power operated~~ isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. 24 The [48] inch purge valves must be maintained ~~sealed~~ closed ~~[or have blocks installed to prevent full opening]. [Blocked purge valves also actuate on an automatic signal.]~~ The valves covered by this LCO are listed ~~along with their associated stroke times~~ in the ~~F~~SAR (Ref. 4). 7

edit

The normally closed, or open under administrative control isolation valves are considered OPERABLE when ~~manual~~ valves are closed, ~~check valves have flow through the valve secured~~, blind flanges are in place, and ~~closed systems are intact~~. These passive isolation valves/devices are ~~those~~ listed in Reference ~~5~~ 4. 40

Blind flanges are addressed within this Specification and treated as a separate type of manual "valve."

edit

~~Purge valves with resilient seals must meet additional leakage rate requirements.~~ The ~~other~~ containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing. 7

This LCO provides assurance that the containment isolation valves ~~and purge valves~~ will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents. 7

(continued)

1
BASES (continued)

34

APPLICABILITY

<Insert B3.6-18A>

~~In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment.~~ In MODES 5 and 6, the probability and consequences of ~~these~~ events are reduced due to the pressure and temperature limitations of these MODES. ~~Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5.~~ The requirements for containment isolation valves during MODE 5 and 6, primarily related to movement of irradiated fuel in the reactor building, are addressed in LCO 3.9.3, "Containment Penetrations."

edit

ACTIONS

7

The ACTIONS are modified by a Note allowing penetration flow paths with inoperable reactor building isolation valves, except for ~~[48] inch~~ purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual ~~operator at the valve controls~~, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls. ~~A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.~~

edit

7

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

edit

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

~~In the event isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry~~

7

(continued)

<INSERT B3.6-18A>

In MODES 1, 2, 3 and 4, the reactor building isolation valves OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES.

BASES

ACTIONS (continued)

into the applicable Conditions and Required Actions of LCO 3.6.1. 7

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable (except for purge valve leakage not within limit), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. 7 38 48

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous

(continued)

Reactor Building

H 1

BASES

ACTIONS

A.1 and A.2 (continued)

92 days" is ~~based on engineering judgment and~~ is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility. edit

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve ~~and a closed system~~ In closed Systems, Condition C provides appropriate actions. edit

Required Action A.2 is modified by ~~a~~ two Note(s) ~~that~~ Note 1 applies to isolation devices located in high radiation areas and allows the devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. <INSERT B3.6-20A> Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small. 18

B.1

With two containment isolation valves in one or more penetration flow paths inoperable ~~(except for purge valve leakage not within limit)~~, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are H 7

(continued)

<INSERT B3.6-20A>

Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

1

BASES

ACTIONS

B.1 (continued)

operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 72 ~~[4]~~ hour Completion Time. The specified time period is reasonable, considering the relative Structural Integrity ~~stability~~ of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure ~~leak tightness of containment and~~ that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

6

edit

edit

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is

6

(continued)

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BASES

ACTIONS

C.1 and C.2 (continued)

written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by ~~a~~ two Note~~s~~ ~~that~~ Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. <INSERT B3.6-22A> Therefore, the probability of misalignment of these devices, once verified to be in the proper position, is small. 18

7

~~D.1, D.2, and D.3~~

~~In the event one or more containment purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a [closed and de-activated automatic valve, closed manual valve, and blind flange]. A purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.~~

~~In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and potentially capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if~~

(continued)

<INSERT B3.6-22A>

Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

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BASES

ACTIONS

D.1, D.2, and D.3 (continued)

7

~~not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.~~

~~For the containment purge valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.6 must be performed at least once every [ ] days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 7). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per [ ] days was chosen and has been shown acceptable based on operating experience.~~

D ~~E~~.1 and D ~~E~~.2

edit

If the Required Actions and associated Completion Times are not met, the unit ~~plant~~ must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit ~~plant~~ must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit ~~plant~~ conditions from full power conditions in an orderly manner and without challenging unit ~~plant~~ systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

7

Each 24 ~~[48]~~ inch containment purge isolation valve in the Purge System supply and exhaust is required to be verified ~~sealed~~ closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

~~sealed~~ closed position during MODES 1, 2, 3, and 4. A containment purge valve that is ~~sealed~~ closed must have motive power to the valve operator removed. This can be ~~accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness.~~ The Frequency is ~~a result of an NRC initiative, Generic Issue B-24 (Ref. 6), related to containment purge valve use during unit operations. In the event purge valve leakage requires entry into Condition D, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.~~

removing the valve handswitch key

consistent with other reactor building isolation valves discussed in SR 3.6.3.2.

edit

7

~~SR 3.6.3.2~~

~~This SR ensures that the minipurge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the minipurge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The minipurge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.~~

7

SR 3.6.3.~~3~~ 2

edit

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable

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(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~3~~ 2 (continued)

edit

of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency ~~is based on engineering judgment and~~ was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves open under administrative controls are not required to meet the SR during the time the valves are open. Insert B3.6-25A

edit

4

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.~~4~~ 3

edit

This SR requires verification that each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "Once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves open under administrative controls are not required to meet the SR during the time they are open. Insert B3.6-25A

4

19

4

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the

(continued)

<Insert B3.6-25A>

This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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BASES

SURVEILLANCE REQUIREMENTS

3

SR 3.6.3.~~4~~ (continued)

edit

probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

4

SR 3.6.3.~~5~~

edit

5

30

Verifying that the isolation time of each ~~power operated and~~ automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period ~~less than or equal to that assumed in the safety analyses~~. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program ~~or 92 days~~.

Power Operated

Consistent with industry standards for sizing valve operators

edit

7

~~SR 3.6.3.6~~

~~For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of once per 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 7).~~

~~Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (greater than that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.~~

5

SR 3.6.3.~~7~~

edit

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~7~~ 5 (continued)

edit

radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The ~~[18]~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ Unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the ~~[18]~~ month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

edit

edit

~~SR 3.6.3.8~~

~~[Reviewer's Note: This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.]~~

~~Verifying that each [48] inch containment purge valve is blocked to restrict opening to ≤ [50%] is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 3 and 4. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.~~

7

7

REFERENCES

~~1. 10 CFR 20.~~ Chapter 5

edit

1 ~~2~~ ~~F~~SAR, ~~Section [6.2]~~. Chapter 14

edit

2 ~~3~~ ~~F~~SAR, ~~Sections [14.1 and 14.2]~~.

3. 10CFR 50.36

28

(continued)

BASES

REFERENCES (continued)	4. ~~F~~SAR, ~~Section [5.3]~~ Table 5-1
	~~5. FSAR, Section [5.3]~~
	~~6. Generic Issue B-24.~~
	~~7. Generic Issue B-20.~~

edit
edit

 1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). ~~These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.~~ 37 14

Insert B3.6-29A

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment ~~functional analyses and the containment structure external pressure analysis~~ DBA and ECCS analyses. Should operation occur outside these limits coincident with a ~~Design Basis Accident~~ (DBA), post accident containment pressures and ECCS performance could exceed calculated values. 37 edit

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB~~, which are analyzed using computer pressure transients~~. The worst-case LOCA generates larger mass and energy release than the worst-case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1). edit

The initial pressure condition used in the containment analysis was ~~[17.7]~~ 14.7 psia ~~([3.0] psig). This resulted in a maximum peak pressure from a LOCA of [53.8] psig.~~ The LCO limit of ~~[3.0]~~ 3.0 psig ensures that, in the event of an accident, the design pressure of ~~[55]~~ 59 psig for containment is not exceeded. ~~In addition, the building was designed for an internal pressure equal to [3] psig above external pressure during a tornado. The containment was also designed for an internal pressure equal to [2.5] psig below external pressure, to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System.~~ The LCO limit of ~~[2.0]~~ -1.0 psig ensures that operation 14 14 37

(continued)

<Insert B3.6-29A>

Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the reactor building design basis accident (DBA) and Emergency Core Cooling System (ECCS) analyses.

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39

14

BASES <INSERT B3.6-30C>

APPLICABLE SAFETY ANALYSES (continued)

within the design ~~limit of [-2.5] psig~~ is maintained (Ref. 2). assumptions for ECCS

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling Systems during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is ~~calculated~~ assumed in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. ~~2~~ 3). edit edit

In MODES 1 and 2, ~~C~~containment pressure satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 4). In MODES 3 and 4, reactor building pressure satisfies Criterion 4 of 10 CFR 50.36. 28

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. ~~Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.~~ 37 14

<Insert B3.6-30A>

34

APPLICABILITY <Insert B3.6-30B>

~~In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within design basis limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.~~ 14

In MODES 5 and 6, the probability and consequences of ~~these~~ events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODES 5 and 6. edit

(continued)

<Insert B3.6-30A>

Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the ECCS analyses.

<Insert B3.6-30B>

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since maintaining reactor building pressure within design basis limits is essential to ensure that the peak reactor building pressure from an accident does not exceed the reactor building design pressure, the LCO is applicable in MODES 1, 2, 3, and 4.

<Insert B3.6-30C>

The LCO limit of 2.0 psig does not consider instrument uncertainty. The LCO limit of -1.0 psig is considered to be an as-indicated value.

1

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If the Required Actions and associated Completion Times are not met

~~If containment pressure cannot be restored within limits within the required Completion Time,~~ the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

edit

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the ECCS and containment analysis (2). The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, ~~including alarms,~~ to alert the operator to an abnormal containment pressure condition.

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REFERENCES

1. FSAR, ~~Section [14.2]~~ Chapter 14.

3 2. 10 CFR 50, Appendix K.

2. SAR, Chapter 5.

4. 10 CFR 50.36.

edit

edit

edit

28

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material, which may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. This LCO ensures that initial conditions assumed in the analysis of a DBA are not violated during unit operations. The total amount of energy to be removed from the Containment Cooling System during post accident conditions is dependent upon the energy released to the containment due to the event as well as the initial containment temperature and pressure. The higher the initial temperature, the higher the resultant peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

12

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses. Average air temperature is also used to establish the containment environmental qualification operating envelope. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment.

Several accidents (primarily LOCA and SLB) result in a marked increase in containment temperature and pressure due to energy release within the containment. Of these, the LOCA results in the greatest sustained increase in containment temperature. By maintaining containment air temperature at less than the initial temperature assumed in

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the LOCA analysis, the reactor building design condition will not be exceeded.

The LOCA that was identified as presenting the greatest challenge to containment OPERABILITY was a cold leg Reactor Coolant System break, of specified size, at a reactor coolant pump suction.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

12

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated, using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

None.

12

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 5 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product(s) radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a unit specific basis).

21 edit
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as discussed in the Safety Analysis Report (SAR), specifically
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edit
27

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and Containment Cooling System provide redundant containment heat removal operation. The Containment Spray System and Containment Cooling System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

Safeguards (ES) edit

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The borated water storage tank (BWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray System

ES edit

(continued)

1

BASES

BACKGROUND

Containment Spray System (continued)

pump suction is manually transferred ~~from the BWST~~ to the containment sump. 21

The Containment Spray System provides a spray of ~~relatively cold~~ borated water ~~mixed with sodium hydroxide from the spray additive tank~~ into the upper regions of containment to ~~reduce the containment pressure and temperature and to reduce the concentration of fission products in the containment atmosphere during a DBA.~~ In the recirculation mode of operation, heat is removed from the containment sump water by the decay heat removal coolers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment heat removal. 33 edit 21

Insert B3.6-36A

The Containment Spray System is actuated automatically by a containment High-High pressure signal ~~coincident with a containment high pressure signal and a low pressure injection signal~~. An automatic actuation opens the Containment Spray System pump discharge valves and starts the ~~two~~ Containment Spray System pumps. ~~A manual actuation of the Containment Spray System requires the operator to actuate two separate switches on the main control board to begin the same sequence.~~ edit edit

Insert B 3.6-36B

~~Containment Cooling System~~

~~The Containment Cooling System consists of three containment cooling trains connected to a common duct suction header with four vertical return air ducts. Each cooling train is equipped with demisters, cooling coils, and an axial flow fan driven by a two speed water cooled electric motor. Each unit connection (two per unit) to the common header is provided with a backpressure damper for isolation purposes.~~ 15 21

~~During normal operation, two containment cooling trains are required to operate. The third unit is on standby and isolated from the operating units by means of the backpressure dampers. The swing unit is equipped with a transfer switch. It can be manually placed to either the "A" or "B" power train to operate in case one of the operating units fails. Upon receipt of an emergency signal, the two operating cooling fans running at high speed will~~

(continued)

<Insert B3.6-36A>

During MODE 1 or 2 the Reactor Building Spray System supports the Spray Additive System function of iodine removal by providing the distribution mechanism. In MODES 3 and 4, sodium hydroxide is not mixed with the spray flow.

<Insert B3.6-36B>

Reactor Building Cooling System

The RB Cooling System during normal operations consists of five (5) chilled water supplied cooling coils each in-line with a fan. Four (4) of these fan and chiller coil circuits have in-line service water cooling coils. During normal operations the service water to these coils is isolated. The post accident configuration of the Reactor Building Cooling System consists of the four service water cooling coils and their respective axial flow fans and dampers arranged as two independent trains.

Upon receipt of an Engineered Safeguards Actuation System (ESAS) RB high pressure signal, the four (4) fans associated with the service water coils receive a start signal, the chilled water is isolated, the service water supply and discharge valves open, the RB cooler bypass dampers open (which causes the return air to bypass the chilled water coils) and the RB cooler backdraft dampers receive an open signal. This equipment is powered from class 1E electrical power.

Each of the four (4) service water coil and fan air paths receives return air separately and directly from the RB atmosphere and discharges through ducting to a common plenum for distribution to the various reactor building spaces. The four (4) fans are mounted vertically on the ventilation units and are axial-flow type. The fan motors are single speed and operate in post-accident conditions at the same speed as normal conditions. Reducing fan motor speed during accident conditions is not required due to the reduced suction pressure drop (and hence fan load relative to normal conditions) created by bypassing the chilled water coils. An RB cooling train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation (Ref. 2). The continuous availability of appropriate service water flow to the RB Cooling System is assured by the periodic addition of a biocide to the Service Water System.

1

BASES

BACKGROUND

Containment Cooling System (continued)

~~automatically stop. The two cooling unit fans connected to the ESF buses will automatically restart and run at low speed, provided normal or emergency power is available.~~

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21

~~In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if they are not already running. If they are running at high (normal) speed, the fans automatically stop and restart in slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher density atmosphere.~~

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break. The postulated DBAs are analyzed, with regard to containment ESF systems, assuming the loss of one ESF bus. This is the worst-case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being inoperable.

reduce

edit
edit
edit

The analysis and evaluation show that, under the worst-case scenario, the highest peak containment pressure is [53.8] psig (experienced during a LOCA). The analysis shows that the peak containment temperature is [276]°F (experienced during a LOCA). Both results are less than the design values. ~~(See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, "Containment Air Temperature," for a detailed discussion.)~~ The analyses and evaluations assume a power level of [2568] MWt, one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of [120]°F and [17.7] psia. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

53.96
283.9
are conservatively reported as 54 psig and 284 °F, respectively, and
12
2568
140
14.7
edit

~~The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a [2.5] psig containment pressure drop and is associated with the sudden cooling effect in the interior of the leak tight~~

37

(continued)

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~containment. Additional discussion is provided in the Bases for LCO 3.6.4.~~ 37

~~The modeled Containment Spray System actuation from the containment analyses is based on a response time associated with exceeding the containment pressure High-High setpoint coincident with a high pressure injection signal to achieve full flow through the containment spray nozzles. The~~ 15
Containment Spray System total ~~response~~ assumed delay time of ~~[56]~~ 300 seconds conservatively envelopes ~~includes~~ diesel generator (DG) startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. ~~2~~ 3). edit edit edit

Containment cooling train ~~performance~~ for post accident conditions is given in Reference ~~3~~ 2. The result of the analysis is that each train can provide ~~33%~~ 100% of the required ~~peak~~ cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, ~~required to perform the accident analyses,~~ is also shown in Reference ~~8~~ 2. edit edit edit

<Insert B3.6-38A>

~~The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the containment pressure high setpoint to achieve full Containment Cooling System air and safety grade cooling water flow.~~ The Containment Cooling System total ~~response~~ assumed delay time of ~~[28]~~ 300 seconds ~~includes signal delay, DG startup (for loss of offsite power), and service water pump startup times (Ref. 3).~~ 15 15

<INSERT B3.3-38C>

In MODES 1 and 2, ~~The Containment Spray~~ System and the Containment Cooling System satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 4). 28

LCO

One reactor building spray train is sufficient to reduce ~~During a DBA, a minimum~~ the combination of one containment cooling train and one containment spray train ~~are required to maintain the containment peak pressure and temperature below the design limits. Additionally, one containment spray train is required~~ during MODE 1 or 2 to ~~remove~~ support the Spray Additive System in the removal of iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, in MODES 1 and 2 two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs. 21 21

<Insert B3.6-38B>

(continued)

<Insert B3.6-38A>

... conservatively envelopes signal delay, DG startup, block loading of equipment, fan startup, and service water pump startup times

<Insert B3.6-38B>

In MODE 3 or 4, one reactor building spray train and one reactor building cooling train are required to be OPERABLE. The LCO is provided with a Note which clarifies this requirement.

<Insert B3.6-38C>

In MODES 3 and 4, the Reactor Building Spray System and Reactor Building Cooling System satisfy Criterion 4 of 10 CFR 50.36.

Reactor Building

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BASES

edit

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LCO (continued)

The ~~Each~~ Containment Spray System ~~typically~~ includes ~~a~~ spray pump~~s~~, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered ~~Safety~~ Safeguards ~~Features~~ Actuation System signal and manually transferring suction to the containment sump.

edit

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The ~~Each~~ Containment Cooling System ~~typically~~ includes ~~an~~ axial flow fans ~~driven demisters~~, cooling coils, dampers, ~~by a two speed water cooled electrical motor~~ single speed fan motors, instruments, and controls to ensure an OPERABLE flow path.

34

APPLICABILITY

Insert B3.6-39A

~~In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment spray trains and containment cooling trains.~~

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

21

With one containment spray train inoperable in MODE 1 or 2, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to ~~perform~~ support the iodine removal and iodine and the containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the ~~Containment Spray System~~ containment spray train OPERABLE, reasonable time for repairs, and the low probability of a DBA occurring during this period.

25

edit

The 10 day portion of the Completion Time for Required Action A.1 is based ~~upon~~ on ~~engineering judgment. It takes into account~~ the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. ~~Refer to Section 1.3, Completion Times, for a more detailed~~

edit

(continued)

<Insert B3.6-39A>

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since an event could cause a release of radioactive material in the reactor building as well as a temperature and pressure rise, the Reactor Building Spray System and the Reactor Building Cooling System are required to be OPERABLE in MODES 1, 2, 3, and 4.

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BASES

ACTIONS

~~A.1 (continued)~~

~~discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.~~

edit

D ~~B~~.1 ~~and B.2~~

If the Required Actions and associated Completion Times are not met

~~If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time~~, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours ~~and to MODE 5 within 84 hours~~. The allowed Completion Time~~s are~~ is, as modified by the Note, reasonable, based on operating experience, to reach the required ~~plant~~ unit conditions from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems. ~~The extended interval to reach MODE 5 allows additional time to attempt restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.~~

21

edit

B ~~C~~.1

With one of the ~~required~~ containment cooling trains inoperable in MODE 1 or 2, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The remaining OPERABLE components ~~in this degraded condition provide iodine removal capabilities and~~ are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

edit 21

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The 10 day portion of the Completion Time for Required Action ~~C~~ B.1 is based ~~u~~on ~~engineering judgment. It takes into account~~ the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. ~~Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.~~

edit

edit

(continued)

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BASES

ACTIONS (continued)

C ~~D~~.1 — 21

With two of the ~~required~~ containment cooling trains inoperable in MODE 1 or 2, one of the ~~required~~ containment cooling trains must be restored to OPERABLE status within 72 hours. The remaining Spray System components ~~in this degraded condition~~ (both spray trains are OPERABLE or else Condition G ~~F~~ is entered) ~~provide~~ support iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an ~~accident~~. The 72 hour Completion Time ~~was developed~~ tak~~ing~~es into account the redundant heat removal capabilities afforded by ~~combinations of~~ the Containment Spray System ~~and Containment Cooling System~~ and the low probability of a DBA occurring during this period.

edit

edit

edit

Insert B3.6-41A

F ~~E~~.1 ~~and E.2~~ — 21

If the Required Action~~s~~ and associated Completion Time~~s~~ of Condition E ~~C or D~~ ~~of this LCO~~ are not met, the unit ~~plant~~ must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit ~~plant~~ must be brought ~~to at least MODE 3 within 6 hours and~~ to MODE 5 within 36 hours. The allowed Completion Time~~s~~ is ~~are~~ reasonable, based on operating experience, to reach the required unit ~~plant~~ conditions ~~from full power conditions~~ in an orderly manner and without challenging unit ~~plant~~ systems.

edit — 21

edit

edit

G ~~F~~.1 — 21

With two containment spray trains inoperable in MODE 1 or 2, or any combination of three or more containment spray and containment cooling trains inoperable, ~~the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.~~

Insert B3.6-41B

SURVEILLANCE REQUIREMENTS

SR 3.6.~~6~~5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in

(continued)

<Insert B3.6-41A>

E.1

With either one required reactor building (RB) spray train or one required reactor building cooling train inoperable in MODE 3 or 4, the inoperable train must be restored to OPERABLE status in 36 hours. The 36 hour Completion Time is reasonable based on consideration of the cooling capacity of the remaining required train of RB cooling or RB spray, the reduced reactor coolant energy in these MODES and the short time spent in these MODES.

<Insert B3.6-41B>

... in MODE 1 or 2, or one required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4, then LCO 3.0.3 must be entered immediately.

The first part of this Condition addresses the loss of Spray Additive System support which would result from two inoperable reactor building spray trains in MODE 1 or 2. The second part of this Condition considers the loss of adequate reactor building cooling capacity in MODE 1 or 2 which would result from the loss of three or more of the four RB spray and RB cooling trains. Finally, the third part of this Condition addresses loss of reactor building cooling capability in MODES 3 and 4 when only one train of RB spray and one train of RB cooling are required.

edit

BASES

edit

SURVEILLANCE REQUIREMENTS

SR 3.6.~~6~~5.1 (continued)

position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, or control room indication that those valves outside containment and capable of potentially being mispositioned are in the correct position.

edit

edit

SR 3.6.~~6~~5.2

Operating each ~~[required]~~ containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. ~~It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action.~~ The 31 day Frequency was developed considering the known reliability of the fan units and controls, the ~~two train~~ redundancy available, and the low probability of a significant degradation of the containment cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

26

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SR 3.6.~~6~~5.3

~~Verifying that each [required] containment cooling train provides an essential raw water cooling flow rate of ≥ [1780] gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 1).~~ Insert B3.6-42A The Frequency was developed considering the known reliability of the ~~Cooling Water~~ System, the ~~two train~~ redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

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SR 3.6.~~6~~5.4

edit

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Verifying that each required containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential

(continued)

<Insert B3.6-42A>

Verifying that a service water flow rate of 1200 gpm is provided to each required reactor building cooling train provides assurance that the original design flow rate is being achieved and that the service water flow rate is not degrading (Ref. 3). Assurance that the flow doesn't degrade by biological fouling between surveillances is provided by the addition of a biocide to the Service Water System whenever the service water temperature is between 60°F and 80°F.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 (continued) 5

pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the Containment Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

measured during; pump; may; edit; edit; edit; edit

SR 3.6.6.5 and SR 3.6.6.6 5 5

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The {18} month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the {18} month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

unit; on; edit; edit; edit; edit

SR 3.6.6.7 5

This SR requires verification that each {required} containment cooling train actuates upon receipt of an actual or simulated actuation signal. The {18} month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the {18} month Frequency.

by control board indication; 5; 5; edit; edit; edit; edit; edit

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~6~~ 5.8

With the containment spray header isolated and drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this Surveillance demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzles, a test at ~~the first refueling and at~~ 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

edit

27

REFERENCES

1. SAR, Section 1.4. ~~10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.~~

edit

~~2.~~ 3. ~~F~~SAR, ~~Section [14.1]~~ Chapter 14.

15

~~3.~~ 2. ~~F~~SAR, ~~Section [6.3]~~ Chapter 6.

edit

~~4. FSAR, Section [14.2].~~

5. ASME, Boiler and Pressure Vessel Code, Section XI.

4. 10CFR50.36.

28

REACTOR BUILDING

B 3.6 ~~CONTAINMENT~~ SYSTEMS

B 3.6.~~7~~ 6 Spray Additive System

BASES

BACKGROUND

reduces

The Spray Additive System ~~is a subsystem of the Containment Spray System that assists in reducing~~ the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

Insert B3.6-45A

edit

1

16

The Containment Spray System and Spray Additive System perform no function during normal operations. In the event of ~~an accident such as~~ a loss of coolant accident (LOCA), ~~however,~~ the Spray Additive System will be automatically actuated upon a ~~high~~ high- containment pressure signal by the Engineered ~~Safety Features~~ Safeguards Actuation System.

edit

edit

16

Insert B3.6-45B

Radioiodine in its various forms is the fission product of primary concern in the evaluation of ~~a DBA~~ the dose consequences of an accident. It is absorbed by ~~the spray~~ a sprayed solution from the containment atmosphere. ~~To enhance the iodine absorption capacity of the spray,~~ the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Sodium hydroxide (NaOH), because of its stability when exposed to radiation and elevated temperature, is the ~~preferred~~ utilized spray additive.

Insert B3.6-45C

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edit

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23

The ~~spray additive~~ NaOH tank is designed and located to permit gravity draining into the Containment Spray System. Both Containment Spray System pumps initially take suction from the borated water storage tank (BWST) via two independent flow paths. The ~~spray additive~~ NaOH tank has a common ~~header~~ outlet that splits and feeds each of the Containment Spray System suction lines. The system is designed to ~~inject~~ discharge at a rate commensurate with the draining rate of the BWST so that all borated water injected is mixed with NaOH.

16

16

The flow rate is proportioned to provide a spray solution with a pH ~~between 7.2 and 11.0~~ which is alkaline (Ref. 1). ~~This~~ The range of alkalinity was established not only to aid in removal of airborne iodine, but also to minimize the corrosion of mechanical system components that would occur if the acidic borated water were not buffered. The pH range also considers the environmental qualification of equipment in containment that may be subjected to the spray.

edit

(continued)

<Insert B3.6-45A>

The Reactor Building Spray System supports the Spray Additive System iodine removal function by providing a distribution mechanism for the solution.

<Insert B3.6-45B>

Actuation of the Spray Additive System opens the sodium hydroxide isolation valves which are powered from independent buses. When the valves are open, the sodium hydroxide solution is ready to be into the RB Spray System headers.

<Insert B3.6-45C>

The sodium hydroxide volume requirement is given in gallons for compatibility with the design analyses. The minimum NaOH tank volume of 9000 gallons preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level.

L6

BASES (continued) reactor building

provides for

1 edit

Edit

APPLICABLE SAFETY ANALYSES

The ~~containment~~ Spray Additive System ~~is essential to~~ the effective removal of airborne iodine within ~~containment~~ the reactor building following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value following the accident. The analysis assumes that most of the containment volume is covered by the spray.

edit

The ~~DBA response~~ delay time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.~~6~~ 5, "Containment Spray and Cooling Systems."

edit

edit

The ~~DBA~~ LOCA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that ~~the entire spray additive tank~~ Sufficient NaOH volume is added to the remaining Containment Spray System flow path.

L6 9 edit

In the evaluation of the worst-case LOCA, the safety analysis assumed that an alkaline containment spray effectively reduced the airborne iodine.

Each Containment Spray System suction line NaOH is equipped with its own gravity feed from the ~~spray additive~~ tank. Therefore, in the event of a single failure within the Spray Additive System (i.e., ~~suction~~ NaOH isolation valve failure), NaOH will still be mixed with the borated water, establishing the alkalinity ~~essential~~ Provide to effective iodine removal.

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The Spray Additive System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2).

28

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and NaOH concentration of the ~~spray additive~~ solution must be sufficient to provide NaOH ~~injection~~ into the spray flow until the Containment Spray System suction path is switched from the BWST to the containment sump and to raise the ~~average spray~~ long term Sump solution pH to a level conducive to iodine removal. The ~~average spray~~ solution pH is ~~between 7.2 and 11.0~~ in the alkaline range. This pH ~~range~~ maximizes the effectiveness of the iodine ~~removal~~ retention mechanism without introducing

9 edit

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edit

edit

edit

(continued)

BASES

LCO (continued)

conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

~~In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.~~ 34 21 1

Insert B 3.6-47A

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6. 21

ACTIONS

A.1

With the containment Spray Additive System spray solution inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment capability of the ~~Containment Spray System~~ for corrosion protection and iodine removal enhancement is reduced or non-existant in this Condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a ~~DBA~~ LOCA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst-case DBA occurring during this period. 16 edit edit

B.1 ~~and B.2~~ 21

~~If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time,~~ If the Required Actions and associated Completion Times are not met, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ must be brought to ~~at least~~ MODE 3 within 6 hours ~~and to MODE 5 within 84 hours~~. The allowed Completion Time ~~are~~ is reasonable, based on operating experience, to reach the required ~~plant~~ conditions from full power conditions in an orderly manner and without challenging ~~plant~~ systems. ~~The extended interval to reach~~ edit 21

(continued)

<Insert B3.6-47A>

In MODES 1 and 2, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Although the core is designed to retain structural integrity during an accident, fuel failure with resultant radioactive material release is postulated and the Spray Additive System is required OPERABLE in MODES 1 and 2.

In MODES 3 and 4, there is no postulated fuel failure contribution to radioactive material release and significantly less need for iodine removal capacity. Also, because of the limited time spent in these MODES, the probability of an event requiring use of the Spray Additive System is low. Therefore, the Spray Additive System is not required to be OPERABLE in MODE 3 or 4.

BASES

ACTIONS

~~B.1 and B.2 (continued)~~

~~MODE 5 allows additional time for restoration of the Spray Additive System and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.~~ 21

SURVEILLANCE REQUIREMENTS

SR 3.6.~~7~~6.1

edit 9

Verifying the correct alignment of ~~spray additive~~ NaOH manual, power operated, and automatic valves in the spray additive system flow path provides assurance that the system is able to provide ~~additive~~ NaOH to the ~~Containment~~ Reactor Building Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside containment capable of potentially being mispositioned are in the correct position. 9 1 9

SR 3.6.~~7~~6.2

edit

edit

To provide the most effective iodine removal, the containment spray ~~must~~ should be an alkaline solution. Since the BWST contents are normally acidic, ~~the volume of~~ the ~~spray additive~~ NaOH tank must provide a sufficient volume of ~~spray additive~~ to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. Insert B 3.6-48A The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected. 9 16 23

(continued)

<Insert B3.6-48A>

The NaOH tank solution minimum volume of 9000 gallons corresponds to a tank level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. This parameter does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The minimum NaOH tank volume preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~7~~ 6.3

This SR provides verification of the NaOH concentration in the ~~spray additive~~ NaOH tank and is sufficient to ensure that the reactor building spray solution being injected into ~~containment~~ is at the correct pH level. The concentration of NaOH in the ~~spray additive~~ tank must be determined by chemical analysis. There is no instrument uncertainty included in the surveillance limit values. Additional allowances for instrument uncertainty are contained in the implementing procedures. The 184 day Frequency is sufficient to ensure that the concentration ~~level~~ of NaOH in the ~~spray additive~~ tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

1 16 edit 9

SR 3.6.~~7~~ 6.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. The ~~[18]~~ 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[18]~~ month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. edit edit edit

~~SR 3.6.7.5~~

~~To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, the flow [rate] in the Spray Additive System is verified once per 5 years. This SR provides assurance that the correct amount of NaOH will be metered into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow [rate].~~ 13

REFERENCES

1. ~~F~~SAR, ~~Section [6.2]~~ Chapter 6. edit

2. 10CFR50.36. 28

1

REACTOR BUILDING

B 3.6 ~~CONTAINMENT~~ SYSTEMS

B 3.6.~~8~~ 7 Hydrogen Recombiners

BASES

BACKGROUND

Permanently installed hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) ~~or steam line break (SLB)~~. The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor ~~is~~ remains in ~~returned to~~ the containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a LOCA ~~Design Basis Accident (DBA)~~. 17 edit

edit

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the control room, a power supply located in the auxillary building, and a recombiner located ~~external to containment~~ in the reactor building. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at approximately 100 scfm ~~3000 cfm at a maximum supply temperature of 120°F~~. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4~~.1~~ volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safeguards (ES) ~~Safety Features~~ bus and is provided with a separate power panel and control panel. 17 17 edit

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than a concentration of 4~~.1~~ v/o following a DBA. This ~~control~~ would prevent a hydrogen burn inside containment, thus ensuring the reactor building pressure and temperature assumed in the accident analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. 17 edit

Hydrogen may accumulate within containment following a LOCA as a result of:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the ~~containment~~ reactor building sump; 1

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to Containment Spray System and Emergency Core Cooling Systems solutions.

edit

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as ~~a~~ has been evaluated function of time following the initiation of the accident ~~is~~ ~~calculated~~. Conservative assumptions ~~recommended~~ presented edit by Reference~~s~~ 1 and 2 are used to maximize the amount of hydrogen calculated. These evaluations demonstrate approximately ~~10~~ 8.9 days are needed for hydrogen concentration to increase to 4.~~0~~ v/o post LOCA without recombiner operation. 17

The hydrogen recombiners satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 3). 28

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst-case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

Insert B3.6-51A

~~In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst-case single failure.~~ 17 34

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced ~~after a LOCA~~ would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an ~~accident~~ event edit

edit

(continued)

<Insert B3.6-51A>

In MODES 1 and 2 the hydrogen recombiner OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Two hydrogen recombiners are required OPERABLE in MODES 1 and 2 to assure control of hydrogen concentration within the reactor building to less than the flammability limit of 4 v/o.

edit

BASES

APPLICABILITY (continued)

requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in a reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA ~~or SLB~~ occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA ~~or SLB~~ (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit. 17

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA ~~or SLB~~ occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA ~~or SLB~~ (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit. 17

~~B.1 and B.2~~

~~[Reviewer's Note: This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.]~~ 10

(continued)

edit

BASES

ACTIONS

B.1 and B.2 (continued)

~~With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by [the containment Hydrogen Purge System/hydrogen recombiner/Hydrogen Ignitor System/Hydrogen Mixing System/Containment Air Dilution System/Containment Inerting System]. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. [Reviewer's Note: The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition: In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability.] [Both] the [initial] verification [and all subsequent verifications] may be performed as an administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.~~

10

~~C.1~~ B.1

If the Required Actions and associated Completion Times are not met

~~If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time,~~ the ~~plant~~ Unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging ~~plant~~ systems.

edit

edit

(continued)

edit

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1 7

edit

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can obtain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and power verified to be ≥ 60 kW. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 18

edit

SR 3.6.8.2 7

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The [18] month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low. 18

edit

SR 3.6.8.3 7

edit

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms. The [18] month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low. 18

17

REFERENCES

1. SAR, Section 6.6.

2 ~~1~~. Regulatory Guide 1.7, Revision ~~1~~ 2.

edit

3. 10CFR 50.36.

28

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
3.7.1	3.7.1	Main Steam Safety Valves (MSSVs)
3.7.2	3.7.2	Main Steam Isolation Valves (MSIVs)
3.7.3	3.7.3	[Main Feedwater Stop Valves (MFSVs), Main Feedwater Control Valves (MFCVs), and Associated Startup Feedwater Control Valves (SFCVs)
3.7.4	N/A	Atmospheric Vent Valves (AVVs)
3.7.5	3.7.5	Emergency Feedwater (EFW) System
3.7.6	3.7.6	Condensate Storage Tank (CST)
3.7.7	N/A	Component Cooling Water (CCW) System
3.7.8	3.7.7	Service Water System (SWS)
3.7.9	3.7.8	Ultimate Heat Sink (UHS)
3.7.10	3.7.9	Control Room Emergency Ventilation System (CREVS)
3.7.11	3.7.10	Control Room Emergency Air Temperature Control System (CREATCS)
3.7.12	3.7.11	Emergency Ventilation System (EVS)
3.7.13	3.7.12	Fuel Handling Pool Ventilation System (FHPVS) .
3.7.14	N/A	Fuel Storage Pool Water Level
3.7.15	3.7.13	Spent Fuel Pool Boron Concentration
3.7.16	3.7.14	Spent Fuel Assembly Storage Area
3.7.17	3.7.4	Secondary Specific Activity
3.7.18	N/A	Steam Generator Level

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Seven MSSVs shall be OPERABLE on each main steam line.

-----NOTE-----

During main steam system hydrotesting in MODE 3, one MSSV is required to be OPERABLE on each main steam line with lift setpoints adjusted to allow testing.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Reduce power in accordance with Table 3.7.1-1. AND A.2 Reduce the nuclear overpower trip setpoint in accordance with Table 3.7.1-1.	4 hours 36 hours
B. Required Action and associated Completion Time not met. OR One or more steam generators with less than two MSSVs OPERABLE.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-----------------------------NOTE------------------------------ 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met during main steam system hydrotesting in MODE 3. --- Verify each required MSSV lift setpoint in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
Allowable Power Level and RPS Nuclear Overpower Trip
Allowable Value versus OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVS OPERABLE (PER SG)	MAXIMUM ALLOWABLE POWER LEVEL (% RTP)	RPS NUCLEAR OVERPOWER TRIP ALLOWABLE VALUE (% RTP)
6	85.7	89.9
5	71.4	74.9
4	57.1	59.9
3	42.8	44.9
2	28.5	29.9
1	14.2	14.9

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSIV(s) inoperable in MODE 1 or 2.	A.1 Restore MSIV(s) to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. --------------NOTE-------------- Separate Condition entry is allowed for each MSIV. ------------------------------------ One or more MSIV(s) inoperable in MODE 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	48 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	-----NOTE----- Only required to be performed in MODES 1 and 2. ---------- Verify isolation time of each MSIV is within the limits specified in the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.7.2.2	-----NOTE----- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. ---------- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs)

LCO 3.7.3 Two MFIVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIV(s) inoperable in MODE 1 or 2.	A.1 Restore MFIV(s) to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. -------------NOTE-------------- Separate Condition entry is allowed for each MFIV. ------------------------------------ One or more MFIV(s) inoperable in MODE 3.	C.1 Close or isolate MFIV. AND C.2 Verify MFIV is closed or isolated.	48 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify the isolation time of each MFIV is within the limits provided in the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.7.3.2	------------------------------NOTES------------------------------ 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. --- Verify that each MFIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be $\leq$ 0.17 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant is $\leq$ 0.17 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

---NOTE---

Only one EFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4 when steam generator is relied upon for heat removal.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One steam supply to turbine driven EFW pump inoperable in MODE 1, 2, or 3.	A.1	Restore steam supply to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One EFW train inoperable for reasons other than Condition A in MODE 1, 2, or 3.	B.1	Restore EFW train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	18 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1	-------------NOTE------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. --------------------------------- Initiate action to restore one EFW train to OPERABLE status.	Immediately
E. Required EFW train inoperable in MODE 4.	E.1	Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	---------------------------NOTE--------------------------- Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching $\geq$ 750 psig in the steam generators. --- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

	SURVEILLANCE	FREQUENCY
SR 3.7.5.3	-----------------------------NOTE----------------------------- Not required to be met in MODE 4 when steam generator is relied upon for heat removal. -- Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4	-----------------------------NOTE----------------------------- Not required to be met in MODE 4 when steam generator is relied upon for heat removal. -- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying manual valve alignment from the "Q" condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days
SR 3.7.5.6	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	18 months

3.7 PLANT SYSTEMS

3.7.6 Q Condensate Storage Tank (QCST)

LCO 3.7.6 The QCST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The QCST inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply. AND A.2 Restore QCST to OPERABLE status.	4 hours AND Once per 12 hours thereafter 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance on steam generator for heat removal.	6 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify QCST volume is $\geq$ 32,300 gallons.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Service Water System (SWS)

LCO 3.7.7 Two SWS loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SWS loop inoperable.	A.1	------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by SWS. 2. Enter Applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal made inoperable by SWS. -------------------------------- Restore SWS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	-----------------------------NOTE----------------------------- Isolation of SWS flow to individual components does not render the SWS inoperable. --	
	Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.8 Emergency Cooling Pond (ECP)

LCO 3.7.8 The ECP shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ECP inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify water level of ECP is $\geq$ 5 ft.	24 hours
SR 3.7.8.2	-----------------------------NOTE----------------------------- Only required to be performed from June 1 through September 30. -- Verify average water temperature at the point of discharge from the ECP is $\leq$ 100°F.	24 hours
SR 3.7.8.3	Verify contained water volume of ECP $\geq$ 70 acre-ft at water level of 5 ft.	12 months

3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.9 Two CREVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVS train inoperable.	A.1 Restore CREVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	C.1 Place OPERABLE CREVS train in emergency recirculation mode. OR C.2. Suspend movement of irradiated fuel assemblies.	Immediately Immediately
D. Two CREVS trains inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREVS trains inoperable during MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each CREVS train for $\geq$ 15 minutes.	31 days
SR 3.7.9.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify the control room isolates and each CREVS train actuates on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Air Conditioning System (CREACS)

LCO 3.7.10 Two CREACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREACS train inoperable.	A.1 Restore CREACS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	C.1 Place OPERABLE CREACS train in operation. OR C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
D. Two CREACS trains inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREACS trains inoperable during MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify each CREACS train starts, operates for at least 1 hour, and maintains control room air temperature $\leq$ 84°F	31 days
SR 3.7.10.2	Verify system flow rate of 9900 cfm $\pm$ 10%	18 months

3.7 PLANT SYSTEMS

3.7.11 Penetration Room Ventilation System (PRVS)

LCO 3.7.11 Two PRVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PRVS train inoperable.	A.1 Restore PRVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met. OR Both PRVS trains inoperable	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each PRVS train for $\geq$ 15 minutes.	31 days
SR 3.7.11.2	Perform required PRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.11.3	Verify each PRVS train actuates on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.12 Fuel Handling Area Ventilation System (FHAVS)

LCO 3.7.12 The FHAVS shall be OPERABLE and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling area.

ACTIONS

-----NOTE-----

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. FHAVS inoperable or not in operation.	A.1 Suspend movement of irradiated fuel assemblies in the fuel handling area.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify FHAVS in operation.	12 hours
SR 3.7.12.2	Perform required FHAVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFPT

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Boron Concentration

LCO 3.7.13 The spent fuel pool boron concentration shall be $\geq$ 1600 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	-------------------NOTE------------------- LCO 3.0.3 is not applicable. --	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the spent fuel pool boron concentration is $\geq$ 1600 ppm.	7 days

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Storage

LCO 3.7.14 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 2 shall be within the acceptable range of Figure 3.7.14-1 or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------ Initiate action to move the noncomplying fuel assembly from Region 2.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.14-1 or Specification 4.3.1.1.	Once prior to storing the fuel assembly in Region 2

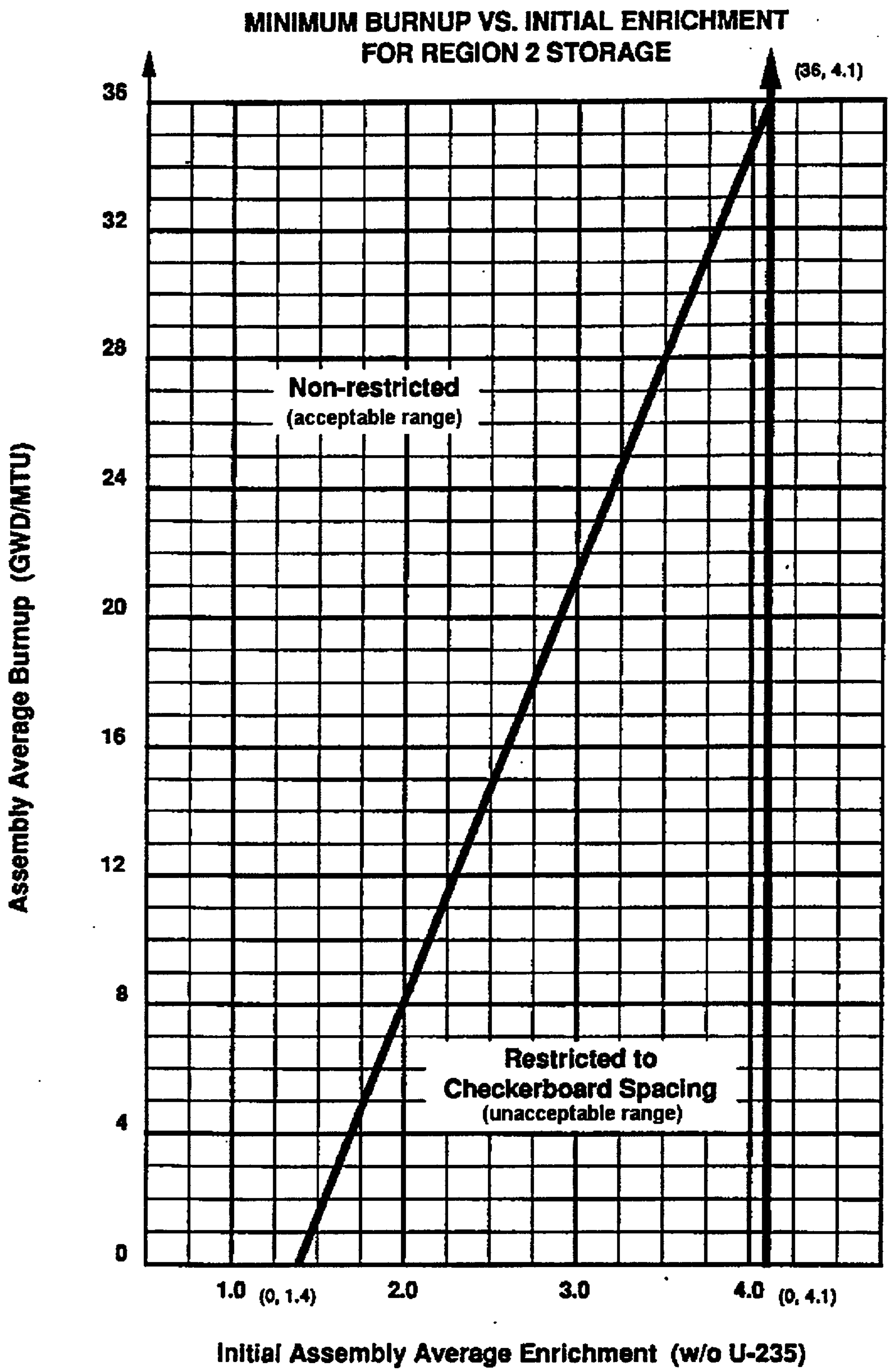


Figure 3.7.14-1 (page 1 of 1)
Burnup versus Enrichment Curve for
Spent Fuel Storage Racks

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Eight MSSVs are located on each main steam header, outside the reactor building, upstream of the main steam isolation valves, as described in the SAR, Section 10.3 (Ref. 1). The MSSV capacity is adequate to meet the requirements of the ASME Code, Section III (Ref. 2). The total capacity of 14 MSSVs is greater than the total steam flow at 102% RTP. The MSSV design includes staggered setpoints (Ref. 1) so that only the needed number of valves will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open the valves.

APPLICABLE SAFETY ANALYSES

The design basis of the MSSVs (Ref. 2) is to limit secondary system pressure to $\leq$ 110% of design pressure when passing 102% of design steam flow (100% plus 2% heat balance error). The MSSVs ensure that the design basis requirements are met for any abnormality or accident considered in the SAR.

The events that may assume use of the MSSVs are those characterized as decreased heat removal events. MSSV use may be assumed during mitigation of the following events:

a. Loss of Load (SAR, Chapter 14 (Ref. 3));

b. Steam generator tube rupture; and

c. Small break loss of coolant (Ref. 3).

The full power turbine trip coincident with a loss of condensate heat sink establishes the required MSSV relief capacity (Ref. 4).

In MODES 1 and 2, the MSSVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODE 3, the MSSVs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The MSSVs are provided to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires fourteen MSSVs (seven on each main steam line) to be OPERABLE to ensure compliance with the ASME Code following DBAs initiated at full power. Operation with less than the required complement of MSSVs requires a limitation on unit THERMAL POWER and adjustment of the Reactor Protection System (RPS) nuclear overpower trip setpoint. The minimum number of OPERABLE MSSVs per steam generator for various power levels and the associated maximum allowable nuclear overpower trip setpoint are identified in Table 3.7.1-1. This effectively limits the Main Steam System steam flow while the MSSV relieving capacity is reduced due to valve inoperability. To be OPERABLE, lift setpoints must remain within limits, according to SR 3.7.1.1.

The safety function of the MSSVs is to open, relieve steam generator overpressure, and reseat when pressure has been reduced.

OPERABILITY of the MSSVs requires periodic surveillance testing in accordance with the Inservice Testing Program.

With all MSSVs OPERABLE, at least one MSSV per steam generator is set at 1050 psig nominal, while the remaining MSSVs per steam generator are set at varied pressures up to and including 1100 psig nominal. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform the design safety function.

The LCO is modified by a Note that allows all but one MSSV on each main steam header to be gagged and the setpoints for the two (one on each header) OPERABLE MSSVs to be reset for the duration of hydrotesting in MODE 3. This is necessary to allow the hydrotest pressure to be attained.

APPLICABILITY

In MODES 1, 2, and 3, the MSSVs are required to be OPERABLE to prevent overpressurization of the main steam system.

In MODES 4 and 5, there is no credible transient requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized. There is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets ASME Code requirements for the power level. Operation may continue, provided the ALLOWABLE THERMAL POWER and RPS nuclear overpower trip setpoint are reduced as required by Table 3.7.1-1. These values are based on the following formulas:

$$RP = \frac{Y}{Z} \times 100\%$$

and

$$SP = \frac{Y}{Z} \times W$$

where:

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1, "Reactor Protection System (RPS)";

Y = Total OPERABLE MSSV relieving capacity per steam generator based on a summation of individual OPERABLE MSSV relief capacities per steam generator (the available capacity of each MSSV is 801,428 lbm/hour);

Z = Required relieving capacity per steam generator of 5,610,000 lbm/hour;

RP = Reduced power requirement (not to exceed RTP); and

SP = Nuclear overpower trip setpoint (not to exceed W).

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, on operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

With one or more steam generators with less than two MSSVs OPERABLE, or if the Required Actions and associated Completion Times are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of MSSV lift setpoints in accordance with the Inservice Testing Program. The safety and relief valve tests are performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6) and include the following for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference 6 provides the activities and frequencies necessary to satisfy the requirements and allows an as-found ± 3% setpoint tolerance. Although not required by the IST Program, the valves are reset to ± 1% during the Surveillance to allow for drift.

This SR is modified by Note 1 that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

The SR is also modified by Note 2 to allow the setpoints for the OPERABLE MSSVs to be reset (one on each header) for the duration of hydrotesting in MODE 3. The remaining valves on each main steam header are typically gagged (or the setpoints may also be reset) to allow the hydrotest pressure to be attained.

REFERENCES

1. SAR, Section 10.3.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. SAR, Chapter 14.
4. Framatome Document 86-1266156-00, "ANO-1 Overpressure Protection," dated October 31, 1997.
5. 10 CFR 50.36.
6. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a main steam line break. MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside of, but close to, the reactor building. The MSIVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Turbine Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam line isolation (MSLI) signal as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." The EFIC System is designed to prevent the simultaneous blowdown of both steam generators. The MSIVs may also be actuated manually.

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the analysis for the steam line break (SLB), as discussed in the SAR, Section 14.2 (Ref. 1). The EFIC System design precludes the blowdown of more than one steam generator, assuming a single active component failure as discussed in the SAR, Section 7.1.4 (Ref. 2).

The SLB outside the reactor building upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The SLB at full power is the limiting case for a post trip return to power. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown.

The MSIVs serve only a closing safety function in the event of an SLB and remain open during power operation.

In MODES 1 and 2, the MSIVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 3, the MSIVs satisfy Criterion 4 of 10 CFR 50.36.

LCO

This LCO requires that the MSIV in each steam line be OPERABLE. For an MSIV to be considered OPERABLE, the isolation time must be within limits and the MSIV must close on an isolation actuation signal when required.

This LCO provides assurance that the MSIVs will perform their design safety function to isolate an SLB.

APPLICABILITY

The MSIVs must be OPERABLE to provide isolation of potential main steam line breaks in MODES 1, 2, and 3, when there is significant mass and energy in the RCS and steam generators.

In MODE 4, the steam generator energy is low. Therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators are depressurized and the MSIVs are not required for isolation of potential main steam line breaks.

ACTIONS

A.1

With one or more MSIVs inoperable in MODE 1 or 2, action must be taken to restore the component to OPERABLE status within 24 hours. Some repairs can be made to the MSIV with the unit hot. The 24 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. Although not credited, the turbine throttle valves may be available to provide isolation for some postulated accidents.

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in MODE 3 within the next 12 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 3.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODE 3, the inoperable MSIV(s) may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the Required Actions and associated Completion Times of Condition C are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 4 within 24 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from MODE 3 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is as specified in the Inservice Testing Program. The MSIV isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

The Frequency for this SR is in accordance with the Inservice Testing Program.

This test is normally conducted in MODE 3, with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The Frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.2.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

REFERENCES

1. SAR, Section 14.2.

2. SAR, Section 7.1.4.

3. 10 CFR 50.36.

4. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs)

BASES

BACKGROUND

The main feedwater isolation valves (MFIVs) isolate main feedwater (MFW) flow to the secondary side of the steam generators. Closing the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside the reactor building and reducing the cooldown effects for SLBs.

The MFIVs close on receipt of a main steam line isolation (MSLI) signal as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." The MFIVs can also be closed manually.

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the SLB.

Failure of an MFIV to close following an SLB, can result in additional mass being delivered to the steam generators, contributing to cooldown.

In MODES 1 and 2, the MFIVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1). In MODE 3, the MFIVs satisfy Criterion 4 of 10 CFR 50.36.

LCO

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators following a main steam line break.

Two MFIVs are required to be OPERABLE. For an MFIV to be considered OPERABLE, the isolation times must be within limits and the MFIV must close on an isolation actuation signal when required.

Failure to meet the LCO requirements can result in a more severe cooldown transient and in additional mass and energy being released to the reactor building following an SLB inside the reactor building.

APPLICABILITY

The MFIVs must be OPERABLE in MODES 1, 2, and 3 to ensure that, in the event of an SLB, the amount of feedwater provided to the affected steam generator is limited. Their closure terminates normal feedwater flow to limit the overcooling transient and to limit the amount of energy that could be added to the reactor building in the case of a secondary system pipe break inside the reactor building.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one or more MFIVs inoperable in MODE 1 or 2, action must be taken to restore the MFIV(s) to OPERABLE status within 24 hours. Some repairs can be made to the MFIV with the unit hot. The 24 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MFIVs occurring during this time interval, and the isolation capability provided by the main feedwater block and control valves. Under normal conditions the main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MFIV problems, thereby avoiding unnecessary shutdown.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in MODE 3 within the next 12 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 3. Further, the unit must be placed in MODE 3 prior to MFIV closure to preclude an undesired transient.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MFIV.

With one or more MFIVs inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 48 hours. The Completion Time is reasonable to attempt minor repairs, and if unsuccessful, to close or isolate the MFIV. Under normal conditions the main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MFIV problems, thereby avoiding unnecessary shutdown. Isolation of an inoperable MFIV may be accomplished by closing the main feedwater block, low load control, and startup control valves (SAR, Table 10-1 (Ref. 2)).

Inoperable MFIVs that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1

If the Required Actions and associated Completion Times of Condition C are not met, the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 4 within 24 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV is as specified in the Inservice Testing Program.

The MFIV isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage. The MFIVs are not tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

This SR verifies that each MFIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.3.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when the steam generator pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

REFERENCES

1. 10 CFR 50.36.

2. SAR, Table 10-1.

B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, abnormalities, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational Leakage") of primary coolant at the limit of 3.5 μCi/gm (LCO 3.4.12, "RCS Specific Activity"). The thyroid dose conversion factors used in the calculation of DOSE EQUIVALENT I-131 are those identified in Section 1.1, "Definitions."

Operating a unit at the allowable limits could result in a 2 hour exclusion area boundary (EAB) exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside the reactor building and a loss of load incident were considered (Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975 (Ref. 2)).

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released (Ref. 2).

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for LCO 3.4.13 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for LCO 3.4.13. For the less probable accident of a steam line break, the assumption is made that a loss of 10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident (Ref. 2).

In MODES 1 and 2, secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, secondary specific activity limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ 0.17 μCi/gm DOSE EQUIVALENT I-131 maintains the radiological consequences of a Design Basis Accident (DBA) significantly less than the Reference 1 guideline doses.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are at low pressure and primary to secondary LEAKAGE is minimal. Therefore, secondary specific activity is not a concern.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis assumptions. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the analysis assumptions are met. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
2. Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975.
3. 10 CFR 50.36

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The EFW pumps take suction from the safety related condensate storage tank (QCST) (LCO 3.7.6, "Q Condensate Storage Tank (QCST)"), and pump to the steam generator secondary side through the EFW nozzles. The core decay heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)"), or atmospheric dump valves (ADVs). If the main condenser is available, steam may be released via the turbine bypass valves.

The EFW System includes one turbine driven EFW pump, and one safety grade motor driven EFW pump. Thus, diversity in motive power sources is provided for the EFW System. The turbine driven EFW pump receives steam from either of the two main steam headers, upstream of the main steam isolation valves (MSIVs).

The EFW System supplies a common header capable of feeding either or both steam generators. Either pump is sufficient to remove decay heat and cool the unit to decay heat removal (DHR) entry conditions. The EFW System initially receives a supply of water from the QCST. The assured safety grade source of water is supplied by the Service Water System (SWS). Valves on the supply piping are manually opened to transfer the water supply from the QCST to the SWS. Water can be supplied from other sources by manually aligning nonsafety grade condensate storage tanks to the EFW pump suction.

The EFW System is capable of supplying feedwater to the steam generators, if required, during normal unit startup and shutdown evolutions, and during hot standby conditions. However, EFW does not provide a normal source of feedwater during these conditions. The normal supplement to the main feedwater system under these conditions is provided by the auxiliary feedwater system.

The EFW actuates automatically (e.g., on loss of main feedwater pumps, low steam generator level, low steam generator pressure, or loss of four reactor coolant pumps) as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation."

The EFW System is discussed in the SAR, Sections 7.1.4 and 10.4.8 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The EFW System is sized to prevent exceeding 110% RCS design pressure for a specified loss of feedwater scenario (Ref. 3).

The design basis of the EFW System is to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure.

The EFW System design is such that it can perform its function with only one EFW train available.

In MODES 1 and 2, the EFW System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODE 3 and MODE 4 when steam generator(s) are relied upon for heat removal, the EFW System satisfies Criterion 4 of 10 CFR 50.36.

LCO

This LCO provides assurance that the EFW System will perform its design function to mitigate the consequences of events that could result in overpressurization of the reactor coolant pressure boundary. Two independent trains are required to be OPERABLE to ensure the availability of residual heat removal capability.

For both EFW trains to be considered OPERABLE, the components and flow paths are required to be capable of providing EFW flow to both steam generators. This requires that the turbine driven EFW pump be OPERABLE with two steam supplies (one from each of the main steam lines upstream of the MSIVs) and capable of supplying EFW flow to the steam generators. The safety grade motor driven EFW pump is also required to be OPERABLE and capable of supplying EFW flow to the steam generators. The piping, valves, instrumentation, and controls in the required flow paths must also be OPERABLE. The primary and secondary sources of water to the EFW System are required to be OPERABLE. The associated flow paths from the EFW System primary and secondary sources of water to both EFW pumps also are required to be OPERABLE.

The LCO is modified by a Note indicating that only one EFW train, which includes the motor driven EFW pump, is required in MODE 4. This is because of reduced heat removal requirement, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE in order to function in the event that the main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

In MODE 4, the EFW System must be OPERABLE when the steam generators are relied upon for decay heat removal since EFW is the safety related source of feedwater to the steam generators. In MODE 4, the steam generators are normally used for heat removal until the DHR System is in operation.

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

ACTIONS

A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, action must be taken to restore the steam supply to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The availability of the redundant OPERABLE steam supply to the turbine driven EFW pump;

b. The availability of the redundant OPERABLE motor driven EFW pump; and

c. The low probability of an event occurring that would require use of the turbine driven EFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required EFW components to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

When one of the required EFW trains (pump or flow path) is inoperable, action must be taken to restore the train to OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven EFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of an event requiring EFW occurring during this time period. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required EFW components to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

With the Required Action and associated Completion Time of Condition A or B not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW train is restored to OPERABLE status.

With both EFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

E.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. Correct alignment for automatic valves may be other than the post-accident position provided the valve is otherwise OPERABLE. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the established acceptance criteria during the cycle. Flow and differential head are indicators of pump performance required by Section XI of the ASME Code (Ref. 5). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 5) satisfies this requirement.

This SR is modified by a Note indicating that the SR may be deferred until suitable test conditions are established. This deferral is required because there may be insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an Emergency Feedwater Initiation and Control (EFIC) System signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Each automatic valve is also verified to be capable of manual operation by over-riding the actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The

18 month Frequency is also acceptable based on operating experience and design reliability of the equipment.

This SR is modified by a Note which states that the SR is not required in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.4

This SR verifies that each EFW pump starts in the event of any accident or transient that generates an EFIC signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This SR is modified by a Note which states that the SR is not required in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.5

This SR ensures that the EFW system is properly aligned by verifying the position of manual valves in the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable in view of other administrative controls, such as SR 3.7.5.1, to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of manual valves has occurred. This SR ensures that the flow path from the QCST to the steam generator is properly aligned.

SR 3.7.5.6

This SR ensures that the EFW flowpath to each steam generator is open and that water reaches the steam generators from the EFW System. This test is performed during shutdown to minimize thermal cycles to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater. The motor-driven EFW pump is specified because of its availability at the low steam generator pressure conditions that exist in the shutdown condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. SAR, Section 7.1.4.
2. SAR, Section 10.4.8.
3. NRC Letter dated January 12, 1981, (1CNA018103).
4. 10 CFR 50.36.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.6 Q Condensate Storage Tank (QCST)

BASES

BACKGROUND

The condensate storage tank (QCST) provides a source of demineralized water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The QCST provides the preferred source of water to the Emergency Feedwater (EFW) System (LCO 3.7.5, "Emergency Feedwater (EFW) System").

Because the QCST is the normally aligned source to EFW, it is designed to withstand earthquakes and other natural phenomena, and a portion is protected from missiles that might be generated by natural phenomena. The QCST is designed as Seismic Category I to ensure availability of the initial EFW supply. Feedwater is also available from alternate sources.

A description of the QCST is found in the SAR, Section 10.4.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The QCST provides the initial source of cooling water to remove decay heat and cool down the unit following any event with a loss of normal feedwater.

A portion of the QCST (T-41B) is protected from tornado generated missiles. The protected volume is sufficient to provide a thirty minute supply of water which is adequate to allow manual operator action, if required, to transfer suction of the EFW pumps to the Service Water System (SWS).

The QCST satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The OPERABILITY of the QCST with the minimum required water volume ensures that sufficient water is available to support EFW operation on both units for at least 30 minutes. This provides adequate time for the operators to manually switch the EFW suction alignment to the Service Water System (SWS), if required. The SWS provides the assured long-term source of cooling water. The required volume considers that the EFW suctions of both units may be aligned to the QCST simultaneously.

The required minimum usable volume includes an allowance for losses due to Unit 2 recirculation line flow. The required volume of 32,300 gallons is equivalent to a tank level of 3 feet 10 inches. This parameter value does not include allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

The tank has sufficient capacity to support more than four hours of cooling in MODE 3 or MODE 4 conditions for both units. This capability is not considered to be a safety related design function and is not controlled by the Technical Specifications.

APPLICABILITY

In MODES 1, 2, 3, and 4 when a steam generator is being relied upon for heat removal, the QCST is required to be OPERABLE.

In MODE 4 when a steam generator is not being relied upon for heat removal, and in MODES 5 and 6, the QCST is not required because the EFW System is not required.

ACTIONS

A.1 and A.2

As an alternative to unit shutdown, the OPERABILITY of the backup water supply should be verified within 4 hours and once every 12 hours thereafter. The OPERABILITY of the backup feedwater supply must include verification, by administrative means, of the OPERABILITY of the flow paths from the backup supply to the EFW pumps and availability of the required volume of water in the backup supply. The QCST must be restored to OPERABLE status within 7 days because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period, requiring the use of water from the QCST.

B.1 and B.2

If the Required Action and associated Completion Times are not met, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generators for heat removal, within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the QCST contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the QCST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in QCST levels.

REFERENCES

1. SAR, Section 10.4.8.
2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.7 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a transient or Design Basis Accident (DBA). During normal operation and normal shutdown, the SWS also provides this function for various safety related and nonsafety related components.

The SWS consists of two independent but interconnected, 100% capacity safety related cooling water loops. Three 100% capacity pumps are provided to supply the two loops. Each loop consists of a pump, piping, valving, sluice gates and instrumentation. The pumps, sluice gates and valves are remote manually aligned. In the unlikely event of a loss of coolant accident (LOCA) essential valves are aligned to their post accident positions upon receipt of an engineered safeguards actuation signal. The SWS provides cooling directly to required plant equipment. The system is also the assured safety related source of water to the emergency feedwater pumps, and can also provide a source of makeup water to the emergency cooling pond, and to the spent fuel pool.

The requirements of the SWS for cooling water are more severe during normal operation (at full power) than under accident conditions. Normal operation requires at least two of the three service water pumps, and the pumps in operation are periodically rotated. Normal operation also includes the addition of a biocide during the reactor building emergency cooler surveillance, when the water temperature is between 60°F and 80°F, to prevent biological fouling of the coolers. This water temperature range provides conditions under which Asian clams can spawn and produce larvae which could pass through SWS strainers.

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the SAR, Section 9.3 (Ref. 1). The principal safety related function of the SWS is the transfer of heat from the reactor and safety related components to the heat sink.

APPLICABLE SAFETY ANALYSES

The primary safety function of the SWS is for one SWS loop, in conjunction with the Low Pressure Injection System and the Reactor Building Cooling System, (reactor building spray, reactor building air coolers, or a combination) to remove core decay heat following a design basis LOCA, as discussed in the SAR, Sections 6.2 and 6.3 (Refs. 2 and 3, respectively).

The SWS is designed to perform its function with a single failure of any active component, assuming loss of offsite power.

The SWS also cools the unit from Decay Heat Removal (DHR) System entry conditions to MODE 5 during normal and post accident operation, as discussed in the SAR, Section 9.5 (Ref. 4). The time required for this evolution is a function of the number of DHR loops that are operating.

The SWS is also required to transfer heat from the diesel generators (DGs).

In MODES 1 and 2, the SWS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 3 and 4, the SWS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two SWS loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

For an SWS loop to be considered OPERABLE, it must have:

a. One OPERABLE pump; and

b. The associated piping, valves, sluice gates, and instrumentation and controls required to perform the safety related function OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS. Therefore, the SWS is required to be OPERABLE in these MODES.

In MODES 5 and 6, the requirements of the SWS are determined by the systems it supports.

ACTIONS

A.1

If one SWS loop is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS loop is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS loop could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable SWS loop results in an inoperable DG. The second Note indicates that the applicable Conditions and

Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable SWS loop results in an inoperable DHR loop. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the Required Action and associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on the existence of procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of components or systems supported by the SWS does not affect the OPERABILITY of the SWS . However, such isolation may render those components inoperable.

SR 3.7.7.2

The SR verifies proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. SAR, Section 9.3.
2. SAR, Section 6.2.
3. SAR, Section 6.3.
4. SAR, Section 9.5.
5. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.8 Emergency Cooling Pond (ECP)

BASES

BACKGROUND

The ECP provides a shared heat sink for removing operating heat from safety related components if the heat sink provided by the Dardanelle Reservoir is unavailable. This is done utilizing the Service Water System (SWS).

The ECP is a portion of the complex of water sources which fulfill the ultimate heat sink requirements for ANO. This complex includes the necessary retaining structures and the piping connecting the sources with, but not including, the SWS intake structure, as discussed in the SAR, Section 9.3 (Ref. 1). The principal function of the ECP is dissipation of residual heat after a reactor shutdown.

The basic performance requirements are that a 30 day supply of water be available for both units, and that the design basis temperatures of safety related equipment not be exceeded. Additional information on the design and operation of the system can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The ECP is the sink for heat removal from the reactor core following an abnormality in which the unit is cooled down and placed on decay heat removal following a loss of the Dardanelle Reservoir inventory which would be considered a single failure.

The operating limits are based on conservative heat transfer analyses for the worst case initial conditions that could be present considering a Unit 2 Design Basis Accident concurrent with a normal shutdown of Unit 1 and a loss of the Dardanelle Reservoir water inventory. Reference 1 provides the details of the assumptions used in the analysis. The minimum ECP requirements take into account: water loss from evaporation due to heat load and climatological conditions, fire pump usage, ECP bottom irregularities, suction pipe level at the ECP, and operator action in transferring the SWS from the Dardanelle Reservoir. Operator action is credited in the inventory analysis during the transfer of the SWS to the ECP. Specifically, pump returns are transferred to the ECP shortly after the Dardanelle Reservoir loss of inventory event begins and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the ECP, lake water is pumped into the ECP, increasing level. This additional water is required, along with that maintained in the ECP, to ensure a 64.5 inch depth, which corresponds to a 30 day supply of cooling water. The ECP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water.

The ECP satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The ECP is a backup system that is required to be OPERABLE to support the SWS. To be considered OPERABLE, the ECP must contain a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 30 days following the design basis event without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the ECP initial temperature should not exceed 100°F, and the volume of water should not fall below 70 acre-feet during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECP is a backup system that is required to support the OPERABILITY of the equipment serviced by the SWS and is required to be OPERABLE in these MODES.

In MODES 5 and 6, the ECP is not required to be OPERABLE.

ACTIONS

A.1 and A.2

If the ECP is inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR (together with SR 3.7.8.3) verifies that adequate long term (30 days) cooling inventory is available. The level specified also ensures NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the ECP level during the applicable MODES. This SR verifies that the ECP indicated water level is ≥ 5 ft.

SR 3.7.8.2

This SR provides assurance that the heat sink for the SWS can dissipate the maximum accident or normal heat loads for 30 days following the design basis event. The measured temperature at the discharge from the ECP is considered a conservative average of total ECP conditions since solar gain, wind speed, and thermal current effects throughout the ECP will essentially be at equilibrium conditions under initial stagnant conditions. The 24 hour Frequency is based on operating experience related to the trending of the ECP temperature during the applicable MODES. This SR verifies that the ECP average water temperature at the point of discharge from the ECP (i.e., SWS suction) is ≤ 100°F.

This SR is modified by a Note indicating that the temperature monitoring is required to be performed only during the summer months (i.e., June 1 to September 30). During other periods of the year, the ECP temperature will not have the potential to reach the temperature limit.

SR 3.7.8.3

This SR (together with SR 3.7.8.1) verifies that adequate inventory exists to support long term (30 days) cooling. The volume specified is verified by soundings to confirm that the indicated level is consistent with the assumed level, and by visual inspections of the physical properties of the ECP. An engineering evaluation is performed of any apparent changes in visual appearance or other abnormal degradation to verify the capability of the ECP to fulfill its safety function. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

REFERENCES

1. SAR, Section 9.3.
2. Regulatory Guide 1.27, Rev. 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974.
3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS is a shared system which provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CREVS consists of two independent, redundant, fan and filter assemblies. Each fan circulates control room air through a filter train consisting of a roughing filter, a high efficiency particulate air (HEPA) filter, and a charcoal adsorber. For control room pressurization, each train provides additional outside air filtered through a four inch bed of charcoal adsorber.

Upon receipt of a unit specific high radiation signal, the control room envelope is isolated, the associated unit's normal control room ventilation system is shutdown, and the associated unit's CREVS is started. The control room envelope is maintained sufficiently leak tight to minimize unfiltered air inleakage. The CREVS operation is discussed in the SAR, Section 9.7 (Ref. 1).

The CREVS is designed to maintain the control room for 30 days of continuous occupancy after a Design Basis Accident (DBA), without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The shared CREVS components are arranged in two safety related ventilation trains, which ensure an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators for the design basis loss of coolant accident fission product release and for a fuel handling accident.

The worst case single active failure of a CREVS component, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

In MODES 1 and 2, and during the movement of irradiated fuel assemblies, the CREVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the CREVS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two CREVS trains are required to be OPERABLE to ensure that at least one is available if a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

For a CREVS train to be considered OPERABLE, the CREVS train must include the associated:

a. OPERABLE fan capable of being powered from both a normal and an OPERABLE emergency power source. (Note: Because this is a shared system, and may be powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREVS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.9 must be applied for inoperable CREVS train(s).);

b. OPERABLE HEPA filter and charcoal adsorber; and

c. OPERABLE ductwork and dampers sufficient to maintain air circulation and provide adequate makeup air flow.

In addition, the control room envelope, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the control room will remain habitable during and following a DBA.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with a release due to a fuel handling accident.

ACTIONS

A.1

With one CREVS train inoperable due to other than the loss of capability for automatic actuation of one fan or one or more isolation dampers in one CREVS train, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room radiation protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1 and B.2

In MODE 1, 2, 3, or 4 if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

During movement of irradiated fuel assemblies, if the Required Action and associated Completion Time of Condition A are not met, the OPERABLE CREVS train must immediately be placed in the emergency recirculation mode. This action ensures that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend movement of irradiated fuel assemblies since this is an activity that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude movement of fuel to a safe position.

D.1

During movement of irradiated fuel assemblies, when two CREVS trains are inoperable, action must be taken immediately to suspend movement of irradiated fuel assemblies since this is an activity that could release radioactivity that could enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude movement of fuel to a safe position.

E.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, the CREVS may not be capable of performing the intended function and a loss of safety function has occurred. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks this system. This test is conducted on alternating trains semi-monthly by initiating flow through the HEPA filters and charcoal adsorbers. The CREVS is designed without heaters and need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and two train redundancy available.

SR 3.7.9.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.9.3

This SR verifies that the control room isolates within 10 seconds, and each CREVS train starts and operates with flow through the HEPA filters and charcoal adsorber banks on an actual or simulated actuation signal. The Frequency of 18 months is consistent with the guidance provided in Regulatory Guide 1.52 (Ref. 3).

REFERENCES

1. SAR, Section 9.7.
2. 10 CFR 50.36.
3. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Air Conditioning System (CREACS)

BASES

BACKGROUND

The CREACS provides temperature control for the control room following isolation of the control room.

The CREACS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, dampers, and instrumentation also form part of the system. During operation, the CREACS maintains the temperature in a range consistent with personnel comfort and long term equipment operation. The CREACS is a subsystem providing air temperature control for the control room.

On detection of high radiation, the control room envelope is isolated, the normal control room ventilation system is shut down, and the CREACS is started. A single train will provide the required temperature control. The CREACS operation to maintain control room temperature is discussed in the SAR, Section 9.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CREACS is to maintain control room temperature for 30 days of continuous occupancy.

The CREACS components are arranged in redundant, safety related trains. A single active failure of a CREACS component does not impair the ability of the system to perform as designed. The CREACS is designed in accordance with Seismic Category I requirements. The CREACS is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements, to ensure a habitable environment and equipment OPERABILITY.

In MODES 1 and 2, and during movement of irradiated fuel assemblies, the CREACS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the CREACS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two independent and redundant trains of the CREACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the control room temperature exceeding limits in the event of an accident.

For a CREACS train to be considered OPERABLE, the individual components that are necessary to maintain control room temperature must be OPERABLE. (Note: Because this is a shared system and is normally powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREACS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.10 must be applied for inoperable CREACS train(s).) These components include the cooling coils, condensing units, and associated temperature control instrumentation. In addition, the CREACS must be capable of maintaining air circulation.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREACS must be OPERABLE to ensure that the control room temperature will not exceed habitability and equipment OPERABILITY requirements following isolation of the control room.

ACTIONS

A.1

With one CREACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CREACS train could result in a loss of CREACS function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining train can provide the required capabilities, and alternate nonsafety related cooling means that are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

C.1 and C.2

During movement of irradiated fuel, if the Required Action and associated Completion Time of Condition A are not met, the OPERABLE CREACS train must be placed in operation immediately. This action ensures that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require the isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of irradiated fuel assemblies, with two CREACS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREACS trains are inoperable in MODE 1, 2, 3, or 4, a loss of safety function has occurred, and LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 and SR 3.7.10.2

These SRs, in conjunction with periodic preventative maintenance activities, provide verification that the CREACS will maintain the control room temperature within acceptable bounds. SR 3.7.10.1 is performed on a staggered basis with one train being tested every two weeks. The Frequencies (31 days and 18 months) are appropriate as periodic preventative maintenance activities are routinely performed and significant degradation of the CREACS is not expected over these time periods.

REFERENCES

1. SAR, Section 9.7.

2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.11 Penetration Room Ventilation System (PRVS)

BASES

BACKGROUND

The PRVS filters air from the penetration areas in the event of penetration leakage from the reactor building during a loss of coolant accident (LOCA).

The PRVS consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, and an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the penetration rooms following receipt of an engineered safeguards actuation system (ESAS) signal.

Following a LOCA, an ESAS signal starts the lead PRVS and if proper flow is not achieved within 20 seconds, the lead system is automatically stopped and 5 seconds later the standby system starts. Upon receipt of the ESAS signal, normal air discharges from the area are isolated, and the air is discharged through the system filters. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PRVS is discussed in the SAR, Sections 6.5 and 14.2.2.5 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the PRVS is established by the LOCA. The system provides filtration for the most likely location of reactor building leakage, i.e., at the penetrations. The analysis of the effects and consequences of a LOCA is presented in Reference 2.

In MODES 1 and 2, the PRVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, the PRVS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two redundant trains of the PRVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power.

For a PRVS train to be considered OPERABLE, its associated:

a. Fan must be OPERABLE;

b. HEPA filter and charcoal adsorber must not be excessively restricting flow, and must be capable of performing their filtration functions; and

c. Required ductwork, and dampers must be OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the OPERABILITY requirements of the reactor building.

In MODES 5 and 6, the PRVS is not required to be OPERABLE since the reactor building is not required to be OPERABLE.

ACTIONS

A.1

With one PRVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the PRVS safety function. However, the overall reliability is reduced because a single failure in the OPERABLE PRVS train could result in loss of PRVS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that of the reactor building (1 hour Completion Time), and this system is not a direct support system for the reactor building. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1 and B.2

If the Required Action and the associated Completion Time are not met, or with both PRVS trains inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. The 31 day Frequency is based on known reliability of equipment and the two train redundancy available.

SR 3.7.11.2

This SR verifies that the required PRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.11.3

This SR verifies that each PRVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with the guidance provided in Regulatory Guide 1.52 (Ref. 4).

REFERENCES

1. SAR, Section 6.5.
2. SAR, Sections 14.2.2.5 and 14.2.2.6.
3. 10 CFR 50.36.
4. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Handling Area Ventilation System (FHAVS)

BASES

BACKGROUND

The FHAVS filters airborne radioactive material from the area of the spent fuel pool following a fuel handling accident.

The FHAVS consists of portions of the normal Auxiliary Building Heating, Ventilation, and Air Conditioning System. The FHAVS consists of a single train which includes a supply fan, prefilter, high efficiency particulate air (HEPA) filter, activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two exhaust fans. Ductwork, dampers, and instrumentation also form part of the system.

During operation, the exhaust from the fuel handling area is passed through the FHAVS exhaust filter and is discharged through the station vent stack.

The FHAVS is discussed in the SAR, Sections 9.7 and 14.2.2 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The FHAVS design basis is established by the fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, credits the FHAVS for a reduction in the amount of airborne radioactive material released to the environment. The assumptions and the analysis are further discussed in Reference 2.

The FHAVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

During movement of irradiated fuel, the FHAVS is required to be OPERABLE and operating.

For the FHAVS to be considered OPERABLE:

1. One exhaust fan must be OPERABLE;

2. HEPA filter and charcoal adsorber must not be excessively restricting flow, and must be capable of performing their filtration functions; and

3. Ductwork and dampers must be OPERABLE.

The FHAVS must be operating since it does not automatically start following a fuel handling accident. A supply fan may be operating, but is not required for FHAVS OPERABILITY.

APPLICABILITY

During movement of irradiated fuel assemblies in the fuel handling area, the FHAVS is always required to be OPERABLE and operating to mitigate the consequences of a fuel handling accident.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note which states that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the FHAVS is inoperable or not in operation during movement of irradiated fuel assemblies in the fuel handling area, immediate action must be taken to preclude the occurrence of an accident. This is achieved by immediately suspending movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Periodic verification of the operation of the FHAVS assures immediate availability of filtration following a fuel handling accident. A 12 hour Frequency is sufficient, considering the system indications and alarms available to the operator for monitoring the FHAVS in the control room.

SR 3.7.12.2

This SR verifies that the required FHAVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. SAR, Section 9.7.
2. SAR, Section 14.2.2.
3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the Bases for LCO 3.7.14, "Spent Fuel Pool Storage," fuel assemblies are stored in the spent fuel pool racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 1600 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron in the spent fuel pool water.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.14-1. Fuel assemblies not meeting the criteria of Figure 3.7.14-1 shall be stored in accordance with Specification 4.3.1.1.e.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978, NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Thus, for accident conditions, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic condition. For example, accident scenarios are postulated which could potentially increase the reactivity and reduce the margin to criticality. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the high density storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.14, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

APPLICABLE SAFETY ANALYSES

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more that eight inches of water separating it from the active fuel in the rest of the rack which precludes interaction). However, accidents can be postulated which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Thus, for accident conditions, the presence of soluble boron in the storage pool water is assumed as a realistic initial condition.

The presence of 1600 ppm boron in the pool water will decrease reactivity by approximately 30% ΔK. Thus $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The specified concentration $\geq$ 1600 ppm of dissolved boron in the spent fuel pool conservatively preserves the assumption used in the analyses of the potential accident scenarios. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movement in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. In addition, action must be immediately initiated to restore the spent fuel pool boron concentration to within its limit. An acceptable alternative is to immediately initiate performance of a spent fuel pool verification to ensure proper locations of the fuel since the last movement of fuel assemblies in the spent fuel pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. Either of these actions are acceptable, and once initiated must be continued until the action is completed. The immediate Completion Time for initiation of these actions reflects the importance of maintaining a controlled environment for irradiated fuel.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SAR, Section 14.2.2.3.

3. Safety Evaluation Report, Section 2.1.3, License Amendment No. 76, April 15, 1983.

4. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel assembly storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The spent fuel pool is sized to store 968 fuel assemblies. The spent fuel storage cells are installed in parallel rows with center to center spacing of 10.65 inches in each direction.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.14-1. Fuel assemblies not meeting the criteria of Figure 3.7.14-1 shall be stored in accordance with paragraph 4.3.1.1.e in SAR Section 4.3, Fuel Storage.

APPLICABLE SAFETY ANALYSES

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies in Region 1. Region 2 controls fuel assembly interaction by fixing the minimum separation between assemblies and by setting enrichment and burnup criterion to limit fissile materials. This is sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to 4.10%. However, fuel assemblies to be stored in the spent fuel pool Region 2 which do not meet enrichment and burnup criterion must be stored in a checkerboard pattern to maintain a k_{eff} of 0.95 or less. In order to prevent inadvertent fuel assembly insertion into two adjacent storage locations, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (unrestricted) are physically blocked before any such fuel assembly is placed in Region 2 (Ref. 1). In addition, the area designated for checkerboard arrangement is divided from the normal storage in Region 2 by a row of vacant storage spaces (Ref. 2).

The spent fuel pool storage satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to Figure 3.7.14-1, ensure that the k_{eff} of the spent fuel pool will always remain ≤ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool. Fuel assemblies not meeting the enrichment and burnup criteria shall be stored in accordance with Specification 4.3.1.1.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.14-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.14-1 or Specification 4.3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.14-1 in the accompanying LCO or Specification 4.3.1.1. For fuel assemblies in the unacceptable range of Figure 3.7.14-1, performance of the SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. SAR, Section 9.6.2.

2. SER for ANO-1 License Amendment No. 76, Section 2.1 (0CNA048314), dated April 15, 1983.

3. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.7: Plant Systems

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 RSTS Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The CTS 4.8.1.b phrase "each EFW flowpath" is clarified to include both the water flow paths and both steam supply flow paths in proposed SR 3.7.5.1. This change is consistent with NUREG-1430.

A4 NUREG 3.7.8 (ITS 3.7.7) Required Action A.1 Notes 1 and 2 are incorporated to retain the CTS cascading inoperability for affected emergency diesel generators and decay heat removal subsystems. Since these would be considered inoperable under the CTS, the addition of these Notes is an administrative change (necessary due to the differing format and implementation of ITS) to retain the CTS requirements. This change is consistent with NUREG-1430.

A5 An explicit Applicability is included for CTS 3.10 as MODES 1, 2, 3, and 4. This is considered equivalent to the CTS even though no explicit applicability is identified with the LCO. The associated Surveillance is identified in CTS Table 4.1-3, item 5, and Notes (7) and (10) identify the applicability for the requirements. In MODES 5 and 6 (CTS cold shutdown and refueling) and when the steam generators are not generating steam (also considered to be cold shutdown and refueling), the secondary coolant is at low temperature and pressure with minimal opportunity for significant release. Therefore, the secondary specific activity is not important. As such, the proposed Applicability is considered equivalent to the current application. This change is consistent with NUREG-1430.

A6 An additional Condition is included for CTS 3.9.1 and 3.9.2 to direct entry into LCO 3.0.3 if both trains of the control room emergency ventilation system (CREVS) or the control room emergency air conditioning system (CREACS) while in MODES 1, 2, 3, or 4. This is equivalent to the CTS requirements and is needed as an explicit condition only due to differences in the implementation. This change is consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

A7 This page is not yet approved in its current form. Therefore, this markup is dependent on the expected NRC approval of the November 23, 1999, license amendment request (LAR) (Ref. 0CAN119906) related to the laboratory testing of activated charcoal, GL 99-02.

A8 The "at greater than 1600 ppm" requirement for boron concentration of the spent fuel pool in CTS 3.8.17 has been revised to "≥ 1600 ppm" in ITS 3.7.13. These are considered to be essentially equivalent since the parameter can be less than the limit, but be so close as to be imperceptible. This change is consistent with design basis and with NUREG-1430.

A9 Not used.

A10 This information has been removed from the ITS since it duplicates requirements provided in the regulations. Such duplication is unnecessary and results in additional administrative burden to revise the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, removal of the duplicative information is considered an administrative change. Further, changes to the requirements are controlled by the NRC. This change is consistent with NUREG-1430.

CTS Location	Duplicated Regulation
3.12.3	10 CFR 30, 40 & 70

A11 Not used.

A12 This page is not yet approved as provided in this package. Therefore, this markup is dependent on the expected NRC approval of the August 6, 1998, (Ref. 1CAN089801) license amendment request (LAR) related to the sodium hydroxide tank level.

A13 Not used.

A14 This page is not yet approved as provided in this package. Therefore, this markup is dependent on the expected NRC approval of the January 27, 2000, (Ref. 0CAN010004) license amendment request (LAR) related to the Q Condensate Storage Tank volume.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 The CTS 3.4.2 shutdown requirements have been revised to adopt the RSTS Completion Times which requires the reactor to be subcritical in 6 hours rather than 12 hours. The RSTS Completion Times also do not allow the additional 48 hours to attempt restoration of compliance. Finally, the RSTS Completion Times for placing the unit in a cold shutdown condition within 12 hours are adopted in lieu of the CTS allowance for an additional 24 hours. These Completion Times are considered to be reasonable and sufficient, considering operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This is considered to be an additional restriction on unit operation which is consistent with NUREG-1430.

M2 The CTS 3.4.1.2 requirements for MSSVs indicate only that 14 of the steam system safety valves are required to be OPERABLE. The CTS does not indicate that these 14 MSSVs must be arranged such that 7 are OPERABLE on the steam line associated with one steam generator and 7 are OPERABLE on the steam line associated with the other steam generator. This specificity is considered to be more restrictive than CTS, but it consistent with the safety analysis and NUREG-1430.

M3 The CTS 3.15.1 requirements are revised to also specifically include a requirement for OPERABILITY of the Fuel Handling Area Ventilation System (FHAVS). Although specific performance criteria are included in the CTS, other ITS requirements for OPERABILITY such as the OPERABILITY of supporting systems could be interpreted as not applicable to the FHAVS. OPERABILITY requirements are appropriate to assure the FHAVS will perform its function when required. This change is considered to be additional restriction on unit operation consistent with NUREG-1430.

M4 The ITS is proposed to contain requirements for periodic verification of the closed status of MSIVs and MFIVs which have been closed as the result of Required Actions. These actions are not currently required since the CTS does not allow continued operation with these valves inoperable, but closed (see related DOCs L3 & L4). These requirements for periodic verification are additional restrictions on unit operation consistent with NUREG-1430.

M5 The CTS requirement (Table 4.1-2, items 13.b & 14.b) to cycle the MSIVs and MFIVs is revised to include the stroke time testing and functional testing of the isolation capability on an actuation signal as normally required for isolation valves. However, since the testing should be accomplished under conditions of operating pressure and temperature and may be required to verify OPERABILITY following work on the valve during a shutdown, a Note is included to allow the testing to be conducted in MODE 3. Allowing testing in MODE 3 (rather than MODE 4, 5, or 6) more closely simulates the conditions under which the valve may be required to perform its safety function. These additional test requirements are considered to be additional restrictions on unit operation consistent with NUREG-1430.

M6 The CTS requirements for OPERABILITY of the Condensate Storage Tank (CST) are expanded to include MODE 4 when the steam generator is relied upon for heat removal. This is consistent with the OPERABILITY requirements for the Emergency Feedwater System and with RSTS LCO 3.7.6. This additional applicability is an additional restriction on unit operation consistent with NUREG-1430.

M7 The CTS 3.4.2 requirements for actions with an inoperable T41B are revised to those presented in NUREG-1430 for the CST. Required Action A.1 has been added which requires the verification by administrative means the operability of the backup water supply (for ANO-1 this is the service water system). This additional Required Action is an additional restriction on unit operation consistent with NUREG-1430.

Additionally, if the CST is not restored to operable status or the backup water supply is not verified to be operable, the Completion Time for placing the unit in a subcritical condition is reduced to 6 hours from 12 hours, and the Completion Time for placing the unit in a condition in which the LCO does not apply after becoming subcritical is reduced from 72 hours to 12 hours. These Completion Times provide sufficient time to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems and are consistent with NUREG-1430.

Finally, a Surveillance Requirement is incorporated to periodically verify the volume of the CST is within limits. The surveillance is necessary to periodically verify the primary EFW water source is available as assumed in the safety analysis. These changes are also additional restrictions on unit operation consistent with NUREG-1430.

M8 An additional Completion Time has been added to those in CTS 3.4.4 to not only require the steam supply to be restored within 7 days from discovery of the inoperable pump (proposed Required Action A.1), or the train within 72 hours (proposed Required Action B.1), but also within 10 days from discovery of failure to meet any of the requirements of the LCO. Currently, for example, if the motor driven pump and one steam supply to the turbine driven pump are concurrently inoperable, separate Actions are entered and the associated Actions are performed with separate Completion Times. Since there are multiple Conditions for different components that are inoperable, it is possible, (however it is extremely unlikely), that the unit can have at least one component inoperable for an unlimited time, and yet a shutdown would never be required (i.e., individual components are repaired within these required restoration times, but there is always at least one component inoperable). The new Completion Time establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO. This is an additional restriction on unit operation consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

M9 CTS 3.3.1 (C) and (I) requires that the service water system pumps and valves be OPERABLE "whenever containment integrity is established as required by Specification 3.6.1." CTS 3.6.1 requires containment integrity whenever RCS pressure is ≥ 300 psig, RCS temperature is ≥ 200°F, and fuel is in the reactor. The ITS requirement for service water pumps is independent of RCS pressure. The pumps and valves will be required with fuel in the reactor and RCS temperature ≥ 200°F. This is an additional restriction on unit operation consistent with NUREG-1430.

M10 CTS 3.3.1(I) requires the valves associated with the service water system to be OPERABLE or locked in the engineered safeguards position, but there are no surveillance requirements specified to verify this requirement. RSTS SR 3.7.8.1 is proposed to be adopted (as ITS SR 3.7.7.1) to periodically verify the position of valves which are not secured in the correct position. ITS SR 3.7.7.1 is also proposed with a Note that indicates that isolation of flow to individual components does not render the SWS inoperable. Overall, this new surveillance is considered an additional restriction on unit operation consistent with NUREG-1430.

M11 CTS 3.3.6 requires that for an inoperable service water subsystem, the unit be placed in a subcritical condition (hot shutdown) within 36 hours of noncompliance, and allows an additional 72 hours to achieve a cold shutdown condition. The ITS provides only 6 hours to achieve MODE 3 (HOT SHUTDOWN) and an additional 30 hours to achieve MODE 5 (COLD SHUTDOWN). The times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This is an additional restriction on unit operation consistent with NUREG-1430.

M12 The CTS functional test of the service water components required by Table 4.1-2, item 9, is expanded to identify more detail as to the content of the test requirements. ITS SR 3.7.7.2 requires each automatic valve that is not secured in its correct post-accident position to be verified to actuate to its correct position on an actual or simulated actuation signal. This additional detail is considered an additional restriction on unit operation consistent with NUREG-1430.

M13 CTS 3.11.1 requires the emergency cooling pond to be OPERABLE whenever containment integrity is established as required by (CTS) Specification 3.6.1. CTS 3.13.1 similarly requires the penetration room ventilation system (PRVS) to be OPERABLE whenever reactor building integrity is required. CTS 3.6.1 requires that reactor building integrity be (established and) maintained whenever all three of the following conditions exist: (a) reactor coolant pressure is 300 psig or greater, (b) reactor coolant temperature is 200°F or greater, and (c) nuclear fuel is in the core. The proposed Applicability for ITS 3.7.8 and ITS 3.7.11 is MODES 1, 2, 3, and 4 which incorporates items (b) and (c) of CTS 3.6.1. However, the ITS requirements will be applicable regardless of reactor coolant pressure. This is an additional restriction on unit operation consistent with NUREG-1430. (For CTS 3.13.1 requirements per CTS 3.6.2, see DOC L17.)

CTS DISCUSSION OF CHANGES

M14 The CTS 3.10 requirement to place the unit in a Hot Standby condition within 6 hours if the secondary specific activity limits are not met is revised to require the unit to be placed in MODE 3 in 6 hours. ITS MODE 3 requires the unit to be subcritical, whereas the CTS Hot Standby required that the unit be at less than 2% of rated power. This is an additional restriction on unit operation consistent with NUREG-1430.

M15 Not used.

M16 The CTS Table 4.1-3, item 4 requirement to perform a spent fuel pool boron concentration verification on a monthly Frequency is revised to a weekly verification (during the times the Specification is applicable; see DOC L15). This is an additional restriction on unit operation consistent with NUREG-1430.

M17 Appropriate Required Actions are incorporated for a condition of noncompliance with CTS 3.8.16 and 3.8.17. The proposed action for CTS 3.8.16 requires the immediate initiation of action to move the noncomplying fuel assembly. The proposed action for CTS 3.8.17 requires prompt restoration of the boron concentration to within limits or removal of the potential for a fuel handling accident. These actions are not explicitly identified in the CTS, and therefore, are additional restrictions on unit operation consistent with NUREG-1430.

M18 CTS 3.9.1 and 3.9.2 contain requirements for OPERABILITY of the CREVS and CREACS during movement of irradiated fuel in the reactor building but does not include an Applicability for movement of irradiated fuel in the fuel handling area, nor does the CTS include ACTIONS for an inoperable train of CREVS or CREACS during these fuel movements. The addition of the Applicability and Required Actions is an additional restriction on unit operation consistent with the safety analysis and with NUREG-1430.

M19 Not used.

M20 An additional intermediate Required Action is added to CTS 3.13.3 to place the unit in MODE 3 within 6 hours if an inoperable penetration room ventilation system (PRVS) train is not restored to OPERABLE status within 7 days. This is an additional restriction on unit operation consistent with NUREG-1430.

M21 Not used.

M22 Not used.

M23 Not used.

CTS DISCUSSION OF CHANGES

M24 NUREG Required Action A.1 is included in ITS 3.7.1 to ensure sufficient MSSV capacity to mitigate an overpressure event. This action is not required in the CTS since continued operation for an indefinite period of time with less than 14 MSSVs is prohibited (see DOC L1). This is an additional restriction on unit operation consistent with NUREG-1430.

NUREG Required Action A.2 is included in ITS 3.7.1 for extra conservatism. Therefore, requirements for reduced maximum allowable nuclear overpower - high trip settings are included based on the number of OPERABLE MSSVs. This is an additional restriction on unit operation consistent with NUREG-1430.

M25 An additional surveillance, beyond CTS 4.17, is included to periodically verify "in operation" as it is required by ITS 3.7.12. This is necessary to verify the assumptions of the safety analysis are met during conditions in which a fuel handling accident may occur. This is an additional restriction on unit operation consistent with NUREG-1430 as modified for unit specific design and analysis. (See also DOD 35.)

M26 CTS 4.8.1.a.1 requires that the turbine driven emergency feedwater pump be tested within 24 hours after reaching the Hot Shutdown condition following a plant heatup and prior to criticality. This is revised in the ITS SR 3.7.5.2 Note to require the testing to be performed with 24 hours after reaching ≥ 750 psig. Since 750 psig occurs prior to reaching 525°F, this test is required to be performed earlier in the startup than it is currently performed. However, the proposed conditions are sufficient to allow the test to be performed and verify OPERABILITY earlier in the conditions applicable to the required equipment. This is an additional restriction on unit operation consistent with NUREG-1430.

M27 CTS 4.8.1.e.2 requires that the automatic actuation of the turbine driven emergency feedwater pump steam supply valves (and the associated turbine driven pump) be tested within 24 hours after reaching the Hot Shutdown condition (if it is not current). This is revised in the ITS SR 3.7.5.3 and SR 3.7.5.4 to require the testing to be performed prior to entry into MODE 3 (i.e., ≤ 280°F). Since the pump is only required to start (and is not required to reach full flow for this test), the test can be performed at less than the 750 psig required for pump flow functional testing. This assures system performance verification occurs prior to entering unit conditions where such performance may be needed to respond to an event. This is an additional restriction on unit operation consistent with NUREG-1430 as modified for unit specific design. (See also DOD 14.)

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 The CTS 3.4.1.2 requirements for 14 OPERABLE MSSVs regardless of the power level have been revised to require only the number of MSSVs required to mitigate an overpressure event initiated at specified power levels. The specific number of MSSVs required for various power levels are identified in ITS Table 3.7.1-1.

CTS 3.4.2 allows operation with less than 14 OPERABLE MSSVs for a period of 24 hours, after which the unit must shutdown. The proposed Required Actions of ITS 3.7.1 Condition A will allow continued operation for an indefinite period of time provided that reactor power is reduced to a level consistent with that provided in Table 3.7.1-1 within 4 hours, and the nuclear overpower trip setpoint is reduced in accordance with Table 3.7.1-1 within 36 hours. These proposed actions will ensure that the relief capacity of the remaining MSSVs is sufficient to mitigate an overpressure event during operation with less than 14 MSSVs OPERABLE. Although this allowance to continue operation beyond 24 hours with less than 14 MSSVs results in a less restrictive requirement, additional restrictions on unit operation (i.e., required power reduction within 4 hours and nuclear overpower trip setpoint reduction within 36 hours) are implemented (See also DOC M24).

L2 The CTS requirement (Table 4.1-2, item 4) for the testing of the MSSV setpoints is revised to allow in-situ testing in MODE 3 during startup. Currently, this testing may be performed either during the pressure and temperature reduction for a shutdown, or during the refueling outage by bench testing. The addition of the Note for ITS SR 3.7.1.1 will allow entry into MODE 3 and testing in MODE 3 during the startup following an outage. This is consistent with current practice at many nuclear power plants and is considered an acceptable method for testing of these valves. This change is consistent with NUREG-1430.

L3 The CTS 3.4.2 requirements for placing the unit in cold shutdown if the other Required Actions are not met is revised to require only that the unit be placed in a condition in which the requirements for the inoperable equipment are not applicable. For the MSSVs, MSIVs, and MFIVs, this will require only that the unit be placed in MODE 4. The CTS required that the unit be placed in cold shutdown (equivalent to ITS MODE 5) even though the equipment was only required above 280°F. This is consistent with NUREG-1430 general application for Required Actions.

CTS DISCUSSION OF CHANGES

L4 The CTS 3.4.2 requirements for shutdown if one MSIV (or one MFIV) is inoperable are proposed to be revised to allow continued operation in MODE 3 if the isolation valve is closed and periodically verified to remain closed. This is appropriate since the only safety function of the isolation valves is closure. The Completion Time is appropriate since the valve isolates a closed system which provides an additional barrier for containment isolation. Therefore, the CTS allowed time for continued operation in MODE 3 prior to any action, i.e., 48 hours, is retained as the proposed Completion Time for isolation valve closure. Since each such inoperability will require an additional closure, a Note is included to allow separate entry into the Condition for each inoperable MSIV (or MFIV). This Note is consistent with NUREG-1430.

L5 The CTS Table 4.1-2 (items 13.a and 14.a) quarterly exercising of the MSIVs and MFIVs is omitted. This exercising, while typically required by Section XI for isolation valves, is normally excepted for MSIVs and MFIVs since even partial stroke testing of these valves increases the risk of a valve closure with the unit generating power. Such a valve closure would result in an unnecessary transient. The normal stroke testing of these valves during startup following a refueling outage (see related DOC M5) provides sufficient verification of the OPERABILITY of these valves. This change is consistent with NUREG-1430.

L6 Not used.

L7 The CTS 3.4.2 requirements for shutdown with an inoperable condensate storage tank (CST) are proposed to be revised to allow continued operation for up to 7 days. Two safety related sources of water are provided for the emergency feedwater (EFW) pumps. The first, and preferred source, is the "Q" CST, T-41B, which is seismically qualified and partially tornado protected. The second, and backup source, is the safety related and seismically qualified service water system. The portion of T-41B which is tornado protected provides a 30 minute supply of water for the EFW pumps which provides time for the operators to manually align the EFW pumps to the alternate source. Since the service water system is required to be OPERABLE (see related DOC M7), the extended Completion Time for an inoperable CST has no significant effect on safety. This 7 day Completion Time is consistent with NUREG-1430.

Additionally, the Required Actions are revised to require the unit to be placed in MODE 4 without reliance on a steam generator for heat removal rather than MODE 5. The proposed action is sufficient to place the unit in a condition which is outside the Applicability of the LCO. This change is consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

L8 The CTS 3.4.4 requirements for actions to be taken with inoperable EFW equipment include requirements for a shutdown with both EFW pumps inoperable if the nonsafety grade auxiliary feedwater (AFW) pump is available. This requirement for a shutdown is proposed to be deleted. While all available documentation may indicate that the AFW pump is available, its actual availability cannot be determined until the unit is partially shutdown to the point that AFW would be placed into service. If AFW is determined to be unavailable at this point, no other source of feedwater is readily available to support continuing the shutdown.

The proposed Required Actions will require that immediate action be taken to restore one EFW pump to OPERABLE status and, if required, initiate shutdown. This proposed action does not remove the normal feedwater system (which is providing feedwater to the steam generators) from service to depend on nonsafety grade equipment for which there is no assurance of availability. This is consistent with the Bases provided for NUREG LCO 3.7.5, Required Action D.1 which states: "the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip." This change is consistent with NUREG-1430.

L9 The CTS 4.8 Surveillance Frequency for EFW pump testing is revised to be consistent with the ASME Section XI requirements. CTS 4.8.1 requires EFW pump testing on a monthly basis. As discussed in NUREG-1366, Section 9.1, industry studies indicate that EFW pump testing on a monthly basis may be contributing to equipment unavailability and that changing the test Frequency to quarterly is reasonably expected to increase the availability of the EFW system. A quarterly Frequency is also consistent with the ASME Section XI requirements. This change is consistent with NUREG-1430 as modified by TSTF-101.

CTS DISCUSSION OF CHANGES

L10 The CTS 4.8 Surveillances are revised to exclude functional requirements for automatic actuation capability to be consistent with the requirements for OPERABILITY of the automatic actuation system. During these excluded operating conditions (i.e., MODE 4), there is more time available for operator action in response to an event which requires emergency feedwater initiation than in higher MODES.

The CTS 4.8 Surveillance Frequency is also revised to exclude that portion of the CTS requirements for performing the turbine driven feedwater pump testing prior to criticality. This is acceptable since the pump is required to be OPERABLE upon entry into the applicable conditions of ITS LCO 3.7.5, and the testing is only a verification of that OPERABILITY. As indicated in Generic Letter 87-09, "it is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed because the vast majority of surveillances do in fact demonstrate that systems or components are OPERABLE." Further, the 24 hours is consistent with the time allowed by SR 3.0.3 to perform the surveillance if it is discovered while in MODE 1 to not have been performed on schedule. This change is consistent with NUREG-1430.

L11 CTS 3.12.2 and 6.12.5.e require that a Special Report be submitted when radioactive material source leakage is identified above certain limits. This report is proposed to be eliminated. This reporting is not required by ITS, and is in addition to the reporting required of other 10 CFR Part 30, Part 40, and Part 70 licensees. The testing for leakage and associated corrective actions, when necessary, are retained under administrative controls (see DOC LA3) but the Special Report is an unnecessary use of licensee and regulator resources since it does not provide a significant corresponding benefit. The reporting criteria of 10 CFR Parts 30, 40, and 70 provide sufficient information. As before, any deficiency which is reportable under 10 CFR Part 30, Part 40, and Part 70, will be reported in accordance with the regulations. This change is consistent with NUREG-1430 and the regulations.

L12 CTS 4.8.1.c is revised to reflect that the verification of manual valve position in each required EFW flow path must be performed prior to entering MODE 2 rather than "prior to relying on the steam generator for heat removal." As discussed in the CTS 4.8.1.c Bases and the Bases for NUREG SR 3.7.5.5, this verification must be made prior to relying on the EFW system for decay heat removal following a subsequent unit shutdown. This change is acceptable because no appreciable change in decay heat magnitude will have occurred during the transition from MODE 5 to MODE 3. This change is consistent with NUREG-1430.

L13 The general CTS 3.3.5 and 3.3.6 requirements which are applicable to an inoperable service water train are revised to be consistent with specific RSTS requirements for an inoperable service water train. CTS 3.3.5 allows a service water train to be made inoperable for up to 24 hours for maintenance, but only if the redundant component in the other train is demonstrated OPERABLE within 24 hours prior to beginning the maintenance. However, the performance of maintenance on one train does not change the basis for believing that the redundant train is OPERABLE, therefore, this requirement is omitted. CTS 3.3.5 is marked as being less restrictive with respect to ITS LCO 3.7.7 because this explicit requirement is not retained in the ITS. The ITS Completion Times are based on the capabilities provided by the OPERABLE train and the low probability of a design basis accident occurring during this time period. This change is consistent with NUREG-1430.

The Completion Time for restoring an inoperable service water train (regardless of the reason for the inoperability) is extended from 36 hours to 72 hours. These Completion Times are based on the capabilities provided by the OPERABLE train and the low probability of a design basis accident occurring during this time period. This change is consistent with NUREG-1430.

L14 The CTS 4.11.5 required time for operation of the penetration room ventilation system (PRVS) is reduced from 1 hour to 15 minutes since the system does not have heaters. Similarly, the CTS 4.17.4 requirements for the FHAVS to operate for at least 10 hours is deleted since the system does not include heaters. Requiring the system to be operated for longer than 15 minutes is unnecessary since the system is required to be in operation during fuel movement. Much longer periods of operation are necessary if the system contains heaters that must operate to periodically dry out the charcoal in the filters. However, this shorter period of operation has been determined to be sufficient for determination that the system functions properly when the system contains no heaters. This change is consistent with NUREG-1430.

L15 The CTS 3.8.17 applicability for spent fuel pool boron concentration has been revised from "at all times" to "When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool." Once fuel assembly movement has ceased and it is verified that there are no misloaded fuel assemblies, there is no further potential for a misloaded fuel assembly or a dropped fuel assembly, either of which could result in a positive reactivity effect which decreases the margin to criticality. Other control of the boron concentration would be for reasons not related to assurance of the results of criticality accident analysis, and therefore, not consistent with the criteria of 10 CFR 50.36 for the content of Technical Specifications. This change is consistent with NUREG-1430.

L16 The CTS 4.10.2.d.2 requirement to test the CREVS actuation with a "control room high radiation test signal" is replaced with the phrase "actual or simulated actuation signal." This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill the surveillance requirements. OPERABILITY is adequately demonstrated in either case since the system can not discriminate between "actual" or "simulated" signals. This change is consistent with NUREG-1430.

L17 CTS 3.13.1 requires the penetration room ventilation system (PRVS) to be OPERABLE whenever reactor building integrity is required. CTS 3.6.2 requires reactor building integrity be (established and) maintained whenever the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown, i.e., enough negative reactivity to remain subcritical by 1% $\Delta k/k$ even with all rods removed and RCS temperature at ~140°F, are not met. The proposed Applicability for ITS 3.7.11 is MODES 1, 2, 3, and 4, and includes no requirements for MODE 6 (refueling shutdown condition), or for MODE 5 with the reactor coolant system otherwise open to the atmosphere.

The PRVS functions to filter reactor building leakage in a post accident environment. In MODE 5 with the reactor coolant system open to the atmosphere, no such accidents are postulated to occur. Therefore, the PRVS function is not required.

ITS 3.9.1 provides requirements for MODE 6 boron concentration. The Required Actions for ITS 3.9.1 provide protection by suspending activities that may initiate an accident and initiating restoration of the required boron concentration. These preventive measures are provided in lieu of actions to provide for mitigation of the event. Typically, the suspension of fuel movement would occur much more rapidly than the reactor building integrity could be established from an unexpected condition. Once there is no potential for an accident, there is no need to require mitigation equipment such as the PRVS. (For CTS 3.13.1 requirements per CTS 3.6.1, see DOC M13.) This change is consistent with NUREG-1430.

L18 Not used.

L19 CTS 3.3.1(I) and 3.3.4(D) require that the engineered safety features valves for the service water system (CTS 3.3.1(C)) be OPERABLE or locked in the Engineered Safeguards (ES) position whenever RB integrity is established and when the reactor is critical. NUREG 3.7.8 requires that the service water system be OPERABLE during MODES 1, 2, 3 and 4. The ES valves, which are components of the service water system, are verified OPERABLE by NUREG SR 3.7.8.2 (which is renumbered and adopted as ITS SR 3.7.7.2). In the NUREG, the ES valves may be verified OPERABLE by actuation to the correct position or by being locked, sealed or otherwise secured in position. These expanded options for ES valve verification will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

CTS Location	New Location
3.4.1.2, Note *	Bases 3.7.1, LCO
3.11.1	Bases 3.7.8, LCO & SR
3.15.2	Bases 3.7.12, RA
4.8.1.e.5	Bases 3.7.5, SR 3.7.5.3
4.10.1.a	Bases 3.7.10, SR 3.7.10.1
4.10.2.a	Bases 3.7.9, SR 3.7.9.1
4.10.2.d.2	Bases 3.7.9, SR 3.7.9.3
4.13.1.3	Bases 3.7.8, SR
4.13.1.4	Bases 3.7.8, SR
5.2.3	Bases 3.7.11, Background

LA2 This information has been moved to the Inservice Testing (IST) Program. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Inservice Testing Program will be controlled by 10 CFR 50.55a and 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
Table 4.1-2, #4	IST Program
Table 4.1-2, #13.b	IST Program
Table 4.1-2, #14.b	IST Program
4.5.1.2.2	IST Program
4.5.2.2.2	IST Program
4.8.1.a	IST Program
4.8.1.d	IST Program

CTS DISCUSSION OF CHANGES

LA3 This information has been moved to the Technical Requirements Manual (TRM) or the Safety Analysis Report (SAR). This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM and the SAR will be controlled by 10 CFR 50.59 and 10 CFR 50.71, as applicable. This change is consistent with NUREG-1430.

CTS Location	New Location
3.3.1.(C)	SAR 9.3.2.1
3.8.16	SAR 9.6.2
Figure 3.8.1	SAR Fig. 9-53
3.12.1	TRM
Table 4.1-3, #4 w/Note (9)	TRM
4.5.1.1.2 (b)	TRM
4.5.2.1.2 (c) (3)	TRM
4.11.5	TRM
4.14	TRM

3.7.7

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

~~Applicability~~

~~Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~ A1

~~Objectivity~~

~~To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~

Specification

3.7.7 Appl. & <LATER> (3.3D, 3.5, 3.6)

3.3.1 The following equipment shall be operable ~~whenever containment integrity is established as required by Specification 3.6.1~~ MODES 1, 2, 3 & 4 M9

<LATER> (3.6)

~~(A) One reactor building spray pump and its associated spray nozzle header.~~ LATER

~~(B) One train of reactor building emergency cooling.~~ loops A1

3.7.7 LCO (C) Two ~~out of three~~ service water pumps shall be operable, ~~powered from independent essential buses, to provide redundant and independent flow paths.~~ LA3 SAR

<LATER> (3.5)

~~(D) Two engineered safety feature actuated Low Pressure Injection (LPI) pumps shall be operable.~~ LATER

3.7.7 LCO (E) ~~Both low pressure injection coolers~~ and their cooling water supplies shall be operable.

<LATER> (3.3D)

~~(F) Two Borated Water Storage Tank (BWST) level instrument channel shall be operable.~~ LATER

<LATER> (3.5)

~~(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.~~ LATER

~~(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote manually operable.~~

3.7.7

<Add SR 3.7.7.1 with Note> M10

L19

3.7.7 LCO & <LATER> (3.5, 3.6)

(I) The engineered safety features valves associated with each of the above systems shall be operable or locked {sealed, or otherwise secured} in the ES position. & LATER

3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350F and irradiated fuel is in the core: <LATER> (3.5) LATER

(A) Two out of three high pressure injection (makeup) pumps shall be maintained operable, powered from independent essential buses, to provide redundant and independent flow paths.

(B) Engineered safety features valves associated with 3.3.2a above shall be operable or locked in the ES position.

3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:

(A) The two core flooding tanks shall each contain an indicated minimum of 13 ± 0.4 feet (1040 ± 30 ft³) of borated water at 600 ± 25 psig.

(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.

(C) The electrically operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.

(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.

3.3.4 The reactor shall not be made critical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable. <LATER> (3.6) LATER

(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.

(B) The sodium hydroxide tank shall contain a volume of ≥9,000 gallons of sodium hydroxide solution at a concentration >5.0 wt% and <16.5 wt%.

(C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be locked open.

A12

3.7.7 LCO

<LATER> (3.5, 3.6)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, sealed, or otherwise secured in the ES position.

L19

LATER

3.3.5 ~~Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~

<LATER> (3.5, 3.6)

L13

LATER

<Add 3.7.7 RA A.1 Notes 1&2> A4

3.7.7 RA A.1 & <LATER> (3.5, 3.6)

~~systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which woul make the affected system train inoperable for more than 24 consecutiv hours. Prior to initiating maintenance on any component of a train i any system, the redundant component of that system shall be demonstra to be operable within 24 hours prior to the maintenance.~~ L13 & LATER

& <LATER> (3.3D, 3.5, 3.6)

3.7.7 RA A.1

3.7.7 RA B.1

3.7.7 RA B.2

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated ~~and the reactor shall be in hot shutdown condition within~~ 36 hours, and, if not corrected, in ~~cold shutdown~~ MODE 5 condition MODE 3 within an additional 72 hours. L13 & LATER A1 M11

<LATER> (3.3D)

~~3.3.7 Exceptions to 3.3.6 shall be as follows:~~

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel sh be operable.~~ LATER

<LATER> (3.5)

~~(B) If the conditions of Specification 3.3.3(D) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.~~ LATER

<LATER> (3.6)

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~ LATER

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of init loss or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~

<LATER> (3.6)

(E) If the conditions of Specification 3.3.4(A) cannot be met becaus one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore the inoper reactor building emergency cooling train to operable status with days of initial loss or be in at least hot shutdown within the n 6 hours and in cold shutdown within the following 30 hours.

LATER

A2

Bases

The requirements of Specification 3.3.1 assure that below 350°F, adequate lo term core cooling is provided. Two low pressure injection pumps are specifi However, only one is necessary to supply emergency coolant to the reactor in event of a loss-of-coolant accident.

The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirem assure that the required post-accident components are available for both rea building emergency cooling and iodine removal. Specification 3.3.1 assures the required equipment is operable.

A train consists of two coolers and their associated fans which have suffici capacity to meet post accident heat removal requirements. Conservatively ea reactor building emergency cooling train consists of two fans powered from t same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.

The borated water storage tank is used for three purposes:

(A) As a supply of borated water for accident conditions.

(B) As an alternate supply of borated water for reaching cold shutdown.(2)

(C) As a supply of borated water for flooding the fuel transfer cana during refueling operation.(3)

3.7.7

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent Δk/k subcritical at 70°F without any control rods in the core. A2

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

A12

A2

REFERENCES

(1) FSAR, Section 14.2.5

(2) FSAR, Section 3.2

(3) FSAR, Section 9.5.2

(4) FSAR, Section 9.3.1

(5) FSAR, Section 6.3

(6) ANO Calculation 91-E-0019-01

3.7.5

3.4.4 If the conditions specified in 3.4.3 cannot be met:

3.7.5 RA E.1 1. With the motor driven EFW pump or its associated flow path inoperable and RCS conditions above CSD and RCS temperature < 280°F and any Steam Generator relied upon for heat removal, immediately initiate action to restore the EFW train to operable status.

3.7.5 RA A.1
3.7.5 RA C.1
3.7.5 RA C.2

2 With the RCS temperature ≥ 280°F and one steam generator supply path to the turbine driven EFW pump inoperable, restore the steam generator supply path to operable status within 7 days or be in Hot Shutdown within 6 hours and reduce RCS temperature to < 280°F within the next 12 hours.

3.7.5 RA B.1
3.7.5 RA C.1
3.7.5 RA C.2

3. With the RCS temperature ≥ 280°F and one EFW pump or its associated flow path inoperable, restore the EFW train to operable status within 72 hours or be in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.

~~4. With the RCS temperature ≥ 280°F, both EFW pumps or their associated flow paths inoperable, and the Auxiliary Feedwater pump available, in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.~~ L8

3.7.5 RA D.1
3.7.5 RA D.1 Note

5. With the RCS temperature ≥ 280°F and both EFW pumps or their associated flow paths inoperable, ~~and the Auxiliary Feedwater pump unavailable,~~ immediately initiate action to restore one EFW train ~~the Auxiliary Feedwater pump~~ to operable status. LCO 3.0.3 and all other LCO Required Actions requiring mode changes are suspended until one EFW train ~~or the Auxiliary Feedwater pump~~ is restored to operable status.

Add 10 day Completion Time for 3.7.5 RA A.1 and RA B.1 M8

3.7.5

A2

Bases

The Emergency Feedwater (EFW) system is designed to provide flow sufficient to remove heat load equal to 3½ percent full power operation. The system minimum flow requirement to the steam generator(s) is 500 gpm. This takes into account a single failure, pump recirculation flow, seal leakage and pump wear.

In the event of loss of main feedwater, feedwater is supplied by the emergency feedwater pumps, one which is powered from an operable emergency bus and one which is powered from an operable steam supply system. Both EFW pumps take suction from tank T41B. Decay heat is removed from a steam generator by steam relief through the turbine bypass, atmospheric dump valves, or safety valves. Fourteen of the steam safety valves will relieve the necessary amount of steam for rated reactor power.

The EFW system is considered to be operable when the components and flow paths required to provide EFW flow to the steam generators are operable. This requires that the turbine driven EFW pump be operable with redundant steam supplies from each of the main steam lines upstream of the MSIVs (CV-2617 and CV-2667) and capable of supplying EFW flow to either of the tv steam generators. The motor driven EFW pump and associated flow path to tl EFW system is also required to be operable. The piping, valves, instrumentation, and controls in the required flow paths shall also be operable. One EFW train, which includes the motor driven EFW pump, is required to be operable when above CSD and below 280°F with any steam generator relied upon for heat removal. This is because of reduced heat removal requirements, the short duration EFW would be required, and the insufficient steam supply available in this condition to power the turbine driven EFW pump.

When one of the required EFW trains is inoperable, action must be taken to restore the train to operable status within 72 hours. This condition includes loss of the steam supply to the turbine driven EFW pump. The 72 hour completion time is reasonable, based on the redundant capabilities afforded by the EFW system, time needed for repairs, and the low probabili: of a DBA occurring during this time period.

With two EFW trains inoperable, the unit must be placed in a mode in which the LCO does not apply using the Auxiliary Feedwater pump. With RCS temperature < 280°F the Decay Heat Removal system may be placed in operatic

With both EFW trains inoperable and the Auxiliary Feedwater pump unavailab: the unit is in a seriously degraded condition with only limited means for conducting a cooldown using nonsafety grade equipment. In such a conditio: the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires tl action be started immediately to restore at least one EFW pump or the Auxi: Feedwater pump to Operable status. LCO 3.0.3 is not applicable, as it cou: force the unit into a less safe condition.

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~ A1

~~Applies to the turbine cycle components for removal of reactor decay heat.~~

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

3.7.1 APPL & <LATER> (3.4A) 3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~ A1 & LATER

~~1. Capability to remove decay heat by use of two steam generators.~~ LATER L1

3.7.1 LCO *2. Fourteen ~~of the steam system~~ safety valves are operable. M2

see pg 40-5 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B. A1

~~4. (Deleted)~~ <ADD TABLE 3.7.1-1> L1

see pg 40-2 & see pg 40-3 5. Both main steam block valves and both main feedwater isolation valves are operable.

Reduce power per Table 3.7.1-1 -- 4hrs M24

3.7.1 RA A.1, 3.7.1 RA B.1, B.2 & <LATER> (3.4A) 3.4.2 ~~Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours,~~ the reactor shall be placed in the ~~hot shutdown~~ MODE 3 condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the ~~cold shutdown condition~~ MODE 4 within 24 hours. L1 A1 M1 L3

See pg 40-4 3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<ADD 3.7.1 ACTIONS Note> L1

<Add 3.7.1 RA A.2> M24

3.7.1 LCO NOTE, SR 3.7.1.1 NOTE 2 * Except ~~that~~ during hydrotests, with the reactor subcritical, ~~fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.~~ LA1 BASES

See pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

LAR

A14

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~ (A1)

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

{MODE 1,2 & 3} (A1) & LATER

3.7.2 APPL & <LATER> (3.4A) 3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~

~~1. Capability to remove decay heat by use of two steam generators.~~ LATER

see pg 40-1 2. Fourteen of the steam system safety valves are operable.

see pg 40-5 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ (A1)

3.7.2 LCO 5. Both main steam block valves and both main feedwater isolation valves are operable.

see pg 40-3

3.7.2 RA A.1, B.1, C.1, D.1 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the ~~hot shutdown~~ MODE 3 (A1) condition within 12 hours. ~~If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.~~ (L4) MODE 4 (L3)

& <LATER> (3.4A)

see pg 40-4 3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<Add 3.7.2 Cond C Note> (L4)

<Add 3.7.2 RA C.2> (M4)

see pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

see pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

(A14)

3.4 STEAM AND POWER CONVERSION SYSTEM

A1

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

MODES 1, 2, & 3

A1 & LATER

3.7.3 APPL & ⟨LATER⟩ (3.4A) 3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~

1. Capability to remove decay heat by use of two steam generators. LATER

See pg 40-1 2. Fourteen of the steam system safety valves are operable.

see pg 40-5 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

A1

~~4. (Deleted)~~

see pg 40-2 3.7.3 LCO 5. Both main steam block valves and both main feedwater isolation valves are operable.

3.7.3 A.1, B.1 C.1, D.1 & ⟨LATER⟩ (3.4A) 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown MODE 3 (A1) condition within 12 hours. ~~If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.~~ (L4) MODE 4 (L3)

See pg 40-4 3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

⟨Add 3.7.3 Cond C Note⟩ L4

⟨Add 3.7.3 RA C.2⟩ M4

See pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

See pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

A14

3.7.5

3.4 STEAM AND POWER CONVERSION SYSTEM

A1

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

See pgs 40-1,2,3,5 — 3.4.1 The reactor shall not be heated above 280°F unless the following conditions are met:

<LATER> (3.4A) — ~~1. Capability to remove decay heat by use of two steam generators.~~ LATER

See pg 40-1 — *2. Fourteen of the steam system safety valves are operable.

See pg 40-5 — 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ A1

See pgs 40-2,3 — 5. Both main steam block valves and both main feedwater isolation valves are operable.

See pgs 40-1,2,3,5 — 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.7.5 LCO 3.4.3 Two (2) EFW trains shall be operable as follows:

3.7.5 APPL Note — 1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

3.7.5 APPL — 2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

See pg 40-1 — * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

SR 3.7.5.2 Note — ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

A14

3.7.6

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~ (A1)

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

MODES 1, 2, 3, & 4 when rely on SG (M6) & LATER

3.7.6 APPL & (LATER) (3.4A) 3.4.1 The reactor shall ~~not~~ be heated above 280°F ~~unless the following conditions are met:~~

1. ~~Capability to remove decay heat by use of two steam generators.~~ LATER

See pg 40-1 2. Fourteen of the steam system safety valves are operable.*

3.7.6 LCO 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ (A1)

See pg 40-2,3 5. Both main steam block valves and both main feedwater isolation valves are operable.

3.7.6 A.2, 3.7.6 B.1, B.2 & (LATER) (3.4A) 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within ~~12~~ 6 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the ~~cold shutdown condition~~ within 24 hours. (L7) & LATER (M7) (L7)

See pg 40-4 3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<Add 3.7.6 RA A.1> (M7)

<Add SR 3.7.6.1> (M7)

See pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

See pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

(A14)

3.7.6

The OPERABILITY of the condensate storage tank with the minimum required water volume ensures that sufficient water is available to support EFW operation for both units for at least 30 minutes. This provides adequate time for the operators to manually switch the EFW suction alignment to the Service Water System (SWS), if required. The SWS provides the assured long-term source of cooling water. The required volume considers that the EFWS of both units may be aligned to T41B simultaneously. The tank is seismically qualified and the required volume is also protected from tornado missiles.

The required minimum usable volume includes an allowance for losses due to Unit 2 recirculation line flow. It does not include any allowance for instrument uncertainty or for the unusable volume due to the suction piping configuration. This volume is equivalent to a tank level of 3'-10".

The tank has sufficient capacity to support more than four hours of cooling flow for both units. This capability is not considered to be a safety related design function and is not controlled by the Technical Specifications.

A2

LAR

414

3.7.2
3.7.3

<LATER> (3.3D) 3.5.1.13 Two control room ventilation radiation monitoring channels shall be operable whenever the reactor coolant system is above the cold shutdown condition or during handling of irradiated fuel. LATER

3.5.1.14 The Main Steam Line Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 10^{-1} to 10^4 mR/hr, whenever the reactor is above the cold shutdown condition. R TRM

<LATER> (3.3C) 3.5.1.15 Initiate functions of the EFIC system which are bypassed at cold shutdown conditions shall have the following minimum operability conditions: LATER

a. "low steam generator pressure" initiate shall be operable when the main steam pressure exceeds 750 psig.

b. "loss of 4 RC pumps" initiate shall be operable when neutron flux exceeds 10% power.

c. "main feedwater pumps tripped" initiate shall be operable when neutron flux exceeds 10% power.

SR 3.7.2.2 Note 2
SR 3.7.3.2 Note 2
<LATER> (3.3C)

3.5.1.16 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig. LATER

3.7.13
3.7.14

<LATER> (4.0) 3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable. LATER

3.7.14 LCO + Appl. SR 3.7.14.1 + <LATER> (4.0) 3.7.14 RA A.1 Note

3.8.16 Storage in Region 2 ~~(as shown on Figure 3.8.1)~~ of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. ~~In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations.~~ The provisions of Specification 3.0.3 are not applicable. LA3 SAR LA3 SAR

3.7.13 LCO + Appl 3.8.17 The boron concentration in the spent fuel pool shall be maintained ~~(at all times)~~ (at greater than 1600 parts per million). L15 A8

3.7.9 LCO + Appl 3.7.10 LCO + Appl 3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable ~~(as required by Specification 3.9)~~. A1

~~**Bases**~~ A2

~~Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.~~

~~The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1)~~

~~The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.~~

~~The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron concentration is sufficient to maintain the core keff ≤ 0.99 if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and~~

<Add 3.7.13 Cond A.> M17

<Add 3.7.14 RA A.1> M17

3.7.13
3.7.14

replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

A2 + R
TRM

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

<LATER>
(3.9)

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface.

LATER

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours (3); and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

<u>REFERENCES</u>

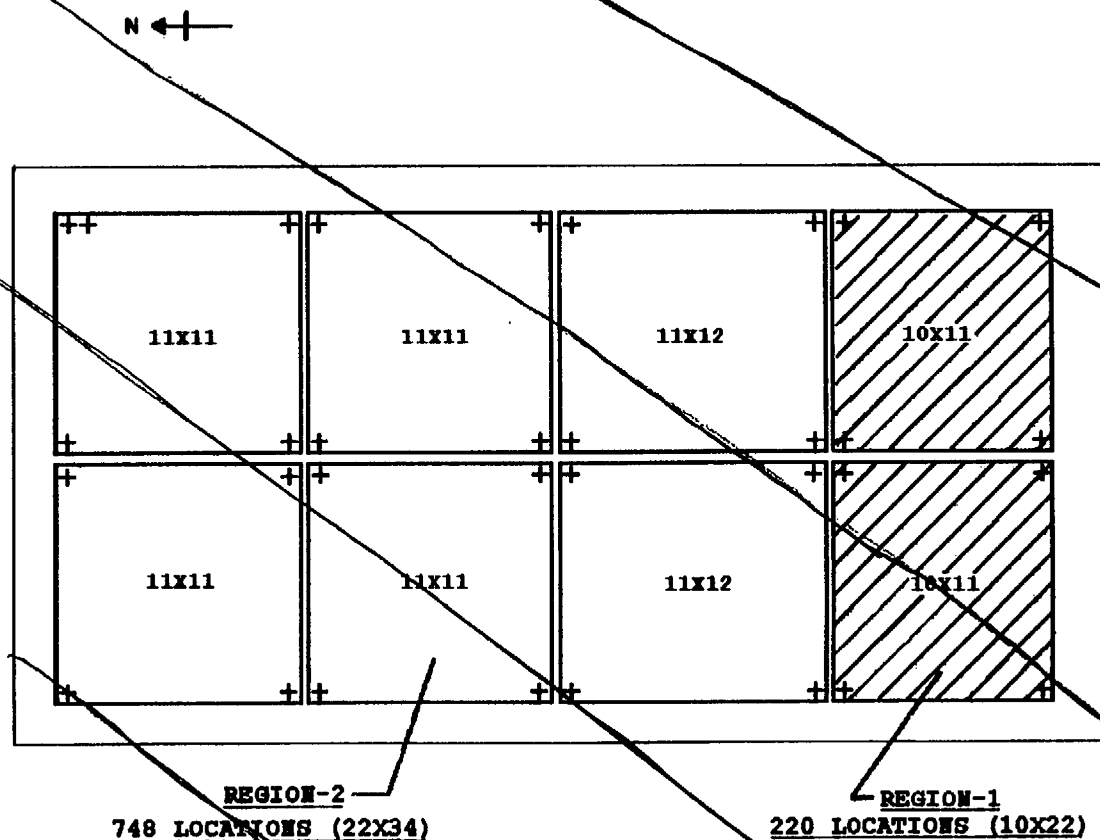
(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

Spent Fuel Pool Arrangement Unit 1

Figure 3.8.1

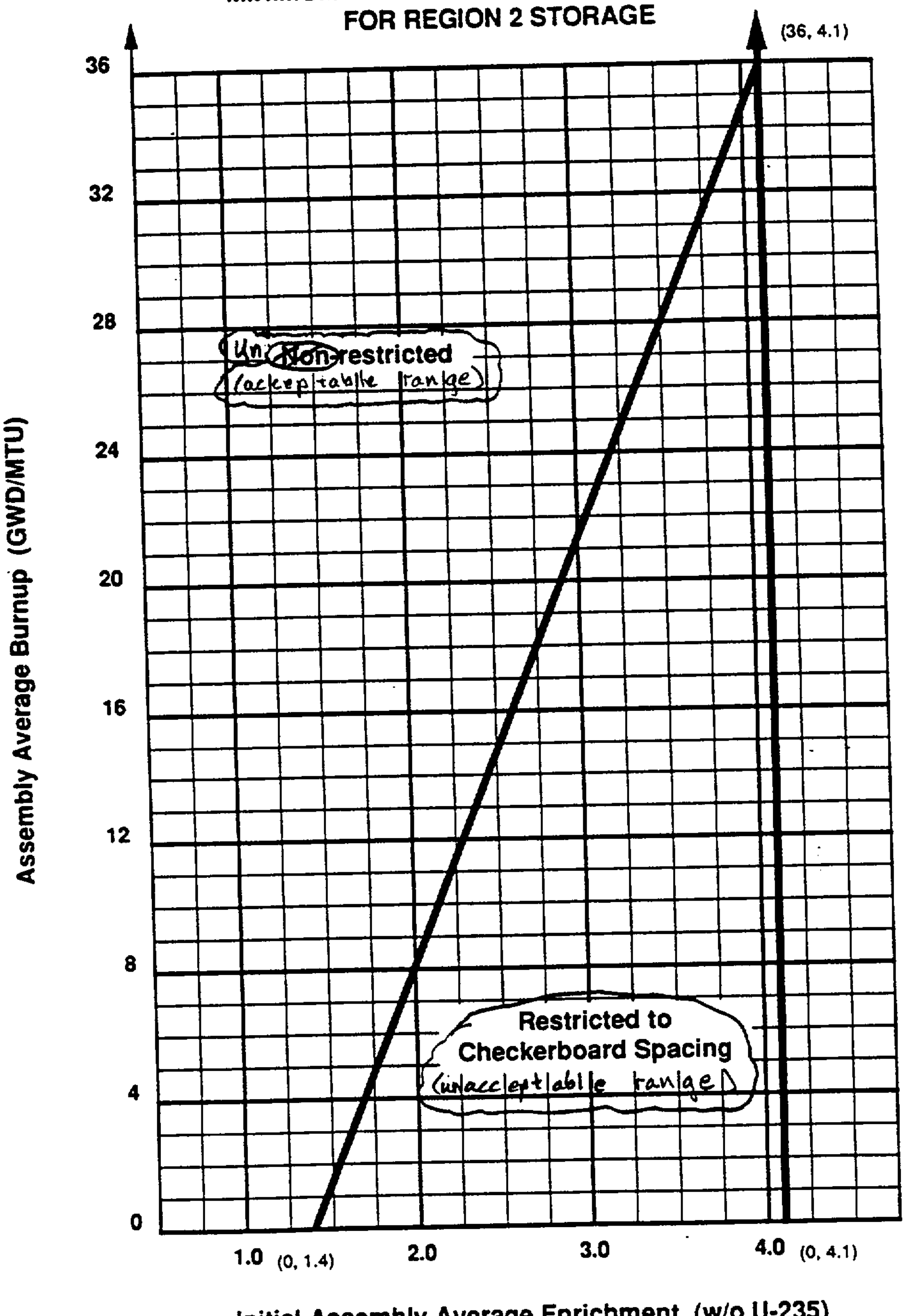


LA's
SAR

3.7.14

F3.7.14-1

~~FIGURE 3.8.2~~
MINIMUM BURNUP VS. INITIAL ENRICHMENT
FOR REGION 2 STORAGE



edit

edit

edit

Figure 3.7.14-1

3.9 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEMS

~~Applicability~~ (A1)

~~Applies to the operability of the control room emergency ventilation and air conditioning systems.~~

~~Objective~~

~~To ensure that the control room emergency ventilation and air conditioning systems will perform within acceptable levels of efficiency and reliability.~~

Specification

3.9.1 Control Room Emergency Air Conditioning System

3.7.10 LCO + Appl.

3.9.1.1 Two independent trains of the control room emergency air conditioning system shall be operable whenever the reactor coolant system is ~~above the cold shutdown condition~~ in MODES 1, 2, 3, or 4 or during handling of irradiated fuel. (A1)

3.7.10 RA A.1
3.7.10 RA B.1
3.7.10 RA B.2

3.9.1.2 With one control room emergency air conditioning system inoperable, restore the inoperable system to Operable status within 30 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.9.2 Control Room Emergency Ventilation System

3.7.9 LCO + Appl.

3.9.2.1 Two independent trains of the control room emergency ventilation system shall be operable whenever the reactor coolant system is ~~above the cold shutdown condition~~ in MODES 1, 2, 3, or 4 or during handling ofirradiated fuel. (A1)

3.7.9 RA A.1
3.7.9 RA B.1, B.2

3.9.2.2 With one control room emergency ventilation system inoperable, restore the inoperable system to Operable status within 7 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

<Add 3.7.9 Conds C & D> (M18)

<Add 3.7.9 Cond E> (A6)

<Add 3.7.10 Conds C+D> (M18)

<Add 3.7.10 Cond E> (A6)

3.7.9
3.7.10

A2

Bases

The control room emergency ventilation and air conditioning system is designed to isolate the combined control rooms to ensure that the control rooms will remain habitable for Operations personnel during and following all credible accident conditions and to ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system. The design configuration of the system is based on limiting the radiation exposure to personnel occupying the control room to 5 REM or less whole body, or its equivalent, in accordance with the requirements of General Design Criteria 19 of Appendix A, 10 CFR 50.

Unit 1 and Unit 2 control rooms are a single environment for emergency ventilation and air conditioning concerns. Since the control room emergency ventilation and air conditioning equipment is shared between units, the plant status of both units must be considered when determining applicability of the specification.

Due to the unique situation of the shared emergency ventilation and air conditioning equipment, the components may be cross fed from the opposite unit per predetermined contingency actions/procedures. Unit 1 may take credit for operability of these systems when configured to achieve separation and independence regardless of normal power and/or service water configuration. This will be in accordance with pre-determined contingency actions/procedures.

The control room emergency ventilation system consists of two independent filter and fan trains, two independent actuation channels and the Control Room isolation dampers. The control room dampers isolate the control room within 10 seconds of receipt of a high radiation signal.

If the actuation signal can not start the emergency ventilation recirculation fan, operating the affected fan in the manual recirculation mode and isolating the control room isolation dampers provides the required design function of the control room emergency ventilation system to isolate the combined control rooms to ensure that the control rooms will remain habitable for operations personnel during and following accident conditions. This contingency action should be put in place immediately (within 1 hour) to fully satisfy the design functions of the control room emergency ventilation system.

The control room emergency air conditioning system (CREACS) provides temperature control for the control room following isolation of the control room. It is manually started from the Unit 2 Control Room. The CREACS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment.

With both trains of the control room emergency ventilation and/or emergency air conditioning inoperable, the function of the control room emergency air systems have been lost, requiring immediate action to place the reactor in a condition where the specification does not apply.

3.7.4

3.10 SECONDARY SYSTEM ACTIVITY

~~Applicability~~

~~Applies to the limiting conditions of secondary system activity for operation of the reactor.~~ A1

~~Objective~~

~~To limit the maximum secondary system activity.~~

Specification

3.7.4 LCO The I-131 dose equivalent of the radioiodine activity in the secondary coolant shall not exceed 0.17 µCi/gm. With the secondary coolant activity

3.7.4 RA A.1 in excess of 0.17 µCi/gm I-131, be in at least Hot Standby MODE 3 within 6 hours M14

3.7.4 RA A.2 and in Cold Shutdown MODE 5 within the following 30 hours.

A1

~~Bases~~

~~For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside containment and a loss of load incident were considered.~~ A2

~~The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released.~~

~~The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for Specification 3.1.4.1 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 µCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for Specification 3.1.4.1. For the less probable accident of a steam line break, the assumption is made that a loss of 1 x 10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 µCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident.~~

3.11 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To assure the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

A1

Specification

MODES 1, 2, 3 & 4

M13

3.7.8 LCO 3.11.1 The emergency cooling pond shall be operable ~~whenever containment integrity is established as required by Specification 3.6.1~~ with:

3.7.8 Appl.

SR 3.7.8.1

SR 3.7.8.2

SR 3.7.8.3

1. A minimum contained water volume of 70 acre-feet (equivalent to an indicated water level of 5 feet).

2. An average water temperature of ≤ 100°F.

LA1

Bases

3.7.8 RA A.1 3.11.2 With the requirements of Specification 3.11.1 not satisfied, be in the hot shutdown condition within 6 hours and in the cold shutdown condition within the following 30 hours.

3.7.8 RA A.2

Bases

The requirements of Specification 3.11.1 provide for sufficient water inventory in the emergency cooling pond to mitigate within acceptable limits the effects of a DBA with a concurrent failure of the Dardanelle Reservoir. The minimum water depth takes into account (1) water loss from evaporation due to heat load and climatological conditions, (2) pond bottom irregularities, (3) suction pipe level at the pond and (4) operator action in transferring the service water system from the Dardanelle Reservoi Operator action is credited in the inventory analysis during the transfer of the service water system to the pond. Specifically, pump returns are transferred to the pond shortly after a loss of lake event and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the pond, lake water is pumped into the pond, increasing level. This additional water is required, along with that maintained by Technical Specifications, to ensure a 64.5 inch pond depth, which corresponds to a 30 day supply of cooling wate

The values are based on worst case initial conditions which could be present considering a simultaneous normal shutdown of Unit 1 and emergency shutdown of Unit 2 following a LOCA in Unit 2, using the ECP as a heat sink. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions.

A2

3.7

LA3
TRM

~~3.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES~~

~~Applicability~~

~~Applies to byproduct, source, and special nuclear radioactive material sources.~~

~~Objective~~

~~To assure that leakage from byproduct, source and special nuclear radioactive material sources does not exceed allowable limits.~~

~~Specification~~

~~3.12.1 The source leakage test performed pursuant to Specification 4.14 shall be capable of detecting the presence of 0.005 μCi of radioactive material on the test sample. If the test reveals the presence of 0.005 μCi or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired, or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 μCi or less of beta and/or gamma emitting material or 5 μCi or less of alpha emitting material. The provisions of Specification 3.0.3 are not applicable.~~

~~3.12.2 A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.12.5 within 90 days if source leakage tests reveal the presence of ≥0.005 microcuries of removable contamination.~~ L11

~~3.12.3 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.~~ A10

3.13 PENETRATION ROOM VENTILATION SYSTEM

Applicability

Applies to the operability of the penetration room ventilation system. (A1)

Objective

To ensure that the penetration room ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.7.11 LCO + Appl

SR 3.7.11.1 2 >

3.13.1 Two independent circuits of the penetration room ventilation system shall be operable ~~whenever reactor building integrity is required~~ with the following performance capabilities: (M13) (L17) (MODES 1, 2, 3 and 4)

<LATER> (5.0) — LATER

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flow (± 10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis from the charcoal adsorber banks shall show the methyl iodide penetration less than 5.0% at velocity within ± 20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

c. Fans shall be shown to operate within ± 10% of design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (± 10%).

e. Air distribution shall be uniform within ± 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system.

SR 3.7.11.3 . f. Each circuit of the system shall be capable of automatic initiation.

3.13.2 If one circuit of the penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days ~~provided that during such seven days all active components of the other circuit shall be operable~~. (A1)

3.7.11 RA A.1

3.13.3 If the requirements of Specifications 3.13.1 and 3.13.2 cannot be met, the reactor shall be placed in ~~the cold shutdown condition~~ (MODE 5) within 36 hours. (A1)

3.7.11 RA B.2

<Add 3.7.11 RA B.1> (M20)

LAR (A7)

A2

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of sealed penetration rooms, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building engineered safety features signal and initially requires no operator action. Each filter train is constructed with a prefilter, a HEPA filter and a charcoal adsorber in series. The design flow rate through each of these filters is 2000 scfm, which is significantly higher than the 1.25 scfm maximum leakage rate from the reactor building at a leak rate of 0.1% per day.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of a least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If one circuit of the penetration room ventilation system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue for a limited period of time while repairs are being made.

LAR A7

3.7.12

3.15 FUEL HANDLING AREA VENTILATION SYSTEM

Applicability

Applies to the operability of the fuel handling area ventilation system. A1

Objective

To ensure that the fuel handling area ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.7.12 LCO + Appl

3.15.1 The fuel handling area ventilation system shall be OPERABLE and in operation whenever irradiated fuel handling operations are in progress in the fuel handling area of the auxiliary building and shall have the following performance capabilities: M3

SR 3.7.12.2

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

<LATER> (5.0) b. The results of laboratory carbon sample analysis shall show the methyl iodide penetration less than 5.0% at a velocity within ±20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%. LATER

c. Fans shall be shown to operate within ±10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%).

e. Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system.

3.7.12 RA A.1

3.7.12 ACTIONS Note

3.15.2 If the requirements of Specification 3.15.1 cannot be met, irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed). The provisions of Specification 3.0.3 are not applicable. LA1 Bases

Bases

The fuel handling area ventilation system is designed to filter the auxiliary building atmosphere during fuel handling operations to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing two exhaust fans and a filter train. The fans are redundant and only one is required to be operating. The filter train consists of a prefilter, a HEPA filter and a charcoal adsorber in series. A2

LAR A7

3.7.12

A2

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine absorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

Allowable Penetration = [100% - methyl iodide efficiency for charcoal credited in accident analysis] / safety factor of 2

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

LAR A7

3.7.1
3.7.7

Table 4.1-2
Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.1)	1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown	LATER
	2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions	
<LATER> (3.4B)	3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month	LATER
SR 3.7.1.1	4. Main Steam Safety Valves	Setpoint <Add SR 3.7.1.1, Note 1>	~~Four Valves Every 18 Mon~~	LA2 TST; L2
	5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown	R TRM
<LATER> (3.4B)	6a. Reactor Coolant System Leakage	Evaluate	Daily	LATER
	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	
	7. Emergency-powered Pressurizer Heaters	Power availability	Daily	
		Heater capacity functional test	Every 18 Months	
<LATER> (3.6)	8. Reactor Building Isolation Trip	Functioning	Every 18 Months	LATER
SR 3.7.7.2	9. Service Water Systems	Functioning	Every 18 Months	M12
	10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool	R TRM

<LATER> (3.1) 1/ Same as tests listed in Section 4.7 — LATER

Notes:

<LATER> (3.4B) — LATER

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

3.7.2
3.7.3

< Add SR 3.7.2.2 & SR 3.7.3.2 with Note 1 with Note 1 > — M5

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

Item	Test	Frequency
<LATER> (3.4B) ~~11. Decay heat removal system isolation valve automatic closure and isolation system~~	~~Functioning~~	~~Each Refueling Shutdown~~ — LATER
<LATER> (5.0) ~~12. Flow limiting annulus on main feedwater line at reactor building penetration~~	~~Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.~~	~~One year, two years, three years, and every five years thereafter measured from date of initial test.~~ — LATER
SR 3.7.2.1 13. Main steam isolation valves <Add Note for SR 3.7.2.1>	~~a. Exercise through approximately 10% travel~~ — M5	~~a. Quarterly~~ — L5
	b. Cycle — M5	~~b. Every 18 months~~ — LA2 IST
SR 3.7.3.1 14. Main feedwater isolation valves <Add Note for SR 3.7.3.1>	~~a. Exercise through approximately 5% travel~~ — M5	~~a. Quarterly~~ — L5
	b. Cycle — M5	~~b. Every 18 months~~ — LA2 IST
<LATER> (3.4A) ~~15. Reactor internals vent valves~~	~~Demonstrate operability by: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).~~	~~Each refueling shutdown~~ — LATER

3.7.4
3.7.13

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
<LATER> (3.4B) 1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
<LATER> (3.1)	d. Dissolved Gases	d. Weekly (7)	LATER
<LATER> (3.4A)	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
<LATER> (3.9)	f. Boron Concentration	f. 3 times/week	LATER & R TRM
<LATER> (3.4B)	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)	LATER
<LATER> (3.5) 2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	M16
SR 3.7.13.1 4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LA3 TRM
SR 3.7.4.1 5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10	R TRM
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	
<LATER> (3.6) 6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	LATER

<LATER> (3.4B) — LATER

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

(2) A radiochemical analysis shall consist of the quantitative measurement the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activitie for the individual isotopes shall be used in the determination of $\bar{E}$. radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activi shall be performed if the measured gross activity changes by more than μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall as given in "Table of Isotopes" (1967) and beta energy per disintegrat shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes. — LATER

<LATER> (3.4B)

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

SR 3.7.4.1 & <LATER> (3.4B)

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity. & LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. — R TRM

<LATER> (3.4B)

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less th 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above. — LATER

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prio to any reactor criticality and analyzed for radioactive iodines of I-1 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

3.7.4 Appl. & <LATER> (3.1 & 3.4B)

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition. — & R TRM & LATER

<LATER> (3.4A)

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition. — LATER

3.7.13 Appl.

(9) Required only when fuel is in the pool ~~and prior to transferring fuel to the pool.~~ — LA3 TRM

3.7.4 Appl.

(10) Not required when not generating steam in the steam generators. — A5 & R TRM

<LATER> (3.4B)

(11) The following shall be required until the end of Cycle 2 operation: — LATER

a. Gross radioiodine shall be determined at least three times per wee during power operation.

3.7.7

4.5 EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

⟨LATER⟩ (3.5)

4.5.1 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirement for emergency core cooling systems.

LATER

Objective

To verify that the emergency core cooling systems are operable.

Specification

4.5.1.1 System Tests

4.5.1.1.1 High Pressure Injection System

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

4.5.1.1.2 Low Pressure Injection System

SR 3.7.7.2 & ⟨LATER⟩ (3.5)

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.

SR 3.7.7.2

(2) Verification of the engineered safeguard function of the service water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers. A1

& ⟨LATER⟩ (3.5)

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel. LA3 TRM & LATER

<LATER>
(3.5)

LATER

4.5.1.1.3 Core Flooding System

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During this test, verification shall be made that the check valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

Approximately quarterly, the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ± 10% of the initial level of performance as determined using test flow paths.

4.5.1.2.2 Valves - Power Operated

(a) At intervals not to exceed three months, each engineered safety feature valve in the emergency core cooling systems and each engineered safety feature valve associated with emergency core cooling in the service water system which are designed to open in the event of a LOCA shall be tested to verify operability.

LA2
IST
& LATER

& <LATER>
(3.5)

(b) The acceptable performance of each power operated valve will be that motion is indicated upon actuation by appropriate signals.

Bases

The emergency core cooling systems are the principle reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

A2

The high pressure injection system under normal operating conditions has one pump operating. At least once per month, operation will be rotated to another high pressure injection pump. This will help verify that the high pressure injection pumps are operable.

The requirements of the service water system for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis will verify that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the borated water storage tank recirc line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

<LATER> (3.6) — (2) Verifying that each operational cooling fan operates for at least 15 minutes. — LATER

SR 3.7.7.2 & <LATER> (3.6) — (c) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the reactor building emergency cooling operation.

SR 3.7.7.2 — (2) Verification of the engineered safety features function of the service water system which supplies the reactor building emergency coolers shall be made to demonstrate operability of the coolers. — A1

SR 3.7.7.2 & <LATER> (3.6) — (3) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly. — LA3 TRM & LATER

<LATER> (3.6) — LATER

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

At intervals not to exceed 3 months the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

4.5.2.2.2 Valves

At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building emergency cooling system and each engineered safety features valve associated with reactor building emergency cooling in the service water system shall be tested to verify that it is operable. — LA2 IST

Bases — A2

The reactor building emergency cooling system and reactor building spray system are redundant to each other in providing post-accident cooling of the reactor building atmosphere to prevent the building pressure from exceeding the design pressure. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building emergency cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable engineered safeguard equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.

3.7.7

A2

Addition of a biocide to service water is performed during reactor building emergency cooler surveillance to prevent buildup of Asian clams in the coolers when service water is pumped through the cooling coils. This is performed when service water temperature is between 60F and 80F since in this water temperature range Asian clams can spawn and produce larva which could pass through service water system strainers.

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or smoke can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the reactor building emergency cooling system are arranged so that they can be visually inspected. The cooling fans and coils and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.

As the reactor building fans are normally operating, starting for testing is unnecessary for those verified to be operating.

Reference

FSAR, Section 6

3.7.5

4.8 EMERGENCY FEEDWATER PUMP TESTING

~~Applicability~~

~~Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.~~

~~Objective~~

~~To verify that the emergency feedwater pump and associated valves are operable.~~

A1

Specification

L9

4.8.1 Each EFW train shall be demonstrated operable:

a) ~~By verifying on a STAGGERED TEST BASIS:~~

M26

SR 3.7.5.2 & Note

1. at ~~least once per 31 days~~ or within 24 hours after reaching the ~~Hot Shutdown condition following a plant heatup and prior to criticality,~~ that the turbine-driven pump ~~starts,~~ operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psig at a flow of ≥ 500 gpm through the test loop flow path.

L10

LA2 IST

SR 3.7.5.2

2. ~~at least once per 31 days~~ by verifying that the motor driven EFW pump ~~starts,~~ operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psi at a flow of ≥ 500 gpm through the test loop flow path.

L9

LA2 IST

SR 3.7.5.1

b) At least once per 31 days by verifying that each valve (manual, power operated or automatic) in each EFW flowpath that is not locked, sealed, or otherwise secured in position is in its correct position.

A3

SR 3.7.5.5

c) Prior to ~~relying upon any steam generator for heat removal~~ <entering MODE 2> whenever the plant has been in ~~CSD or less~~ MODE 5, MODE 6, or defueled for > 30 days, verify proper alignment of each manual valve in each required EFW flow path, which if mispositioned may degrade EFW operation, from the 'Q' condensate storage tank to each steam generator.

L12

A1

d) ~~At least once per 92 days by cycling each motor-operated valve in each flowpath through at least one complete cycle.~~

LA2 IST

e) At least once per 18 months by functionally testing each EFW train and:

SR 3.7.5.3

1. Verifying that each automatic valve in each flowpath actuates automatically to its correct position on receipt of an actual or simulated actuation signal.

<Add SR 3.7.5.3, Note> L10

<Add SR 3.7.5.4, Note> L10

SR 3.7.5.3
SR. 3.7.5.4

2. Verifying that the automatic steam supply valves associated with the steam turbine driven EFW pump actuate to their correct positions upon receipt of an actual or simulated actuation signal. ~~This test is not required to be performed until 24 hours after reaching the Hot Shutdown condition.~~ M27

SR 3.7.5.4

3. Verifying that the motor-driven EFW pump starts automatically upon receipt of an actual or simulated actuation signal.

SR 3.7.5.6

4. Verifying that feedwater is delivered to each steam generator using the electric motor-driven EFW pump.

SR 3.7.5.3

5. Verifying that the EFW system ~~can be operated manually by over-riding automatic signals to the EFW~~ valves. LA1 Bases

A2

~~Bases~~

~~The monthly testing frequency will be sufficient to verify that both emergency feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps. The cycling of the emergency valves assures valve operability when called upon to function. Testing of the turbine dri[illegible] EFW pump is delayed until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perfor[illegible] the test at 280°F. Testing may occur at a lower steam generator pressure if operational experience shows that sufficient steam pressure to perform the [illegible] exists.~~

~~Surveillance Requirement 4.8.1.c ensures that the EFW system is properly al[illegible] by verifying the flow paths to each steam generator prior to relying upon a[illegible] steam generator for heat removal after more than 30 days in Cold Shutdown o[illegible] below. Operability of the EFW flow paths must be demonstrated before suffi[illegible] core heat is generated that would require the operation of the EFW system [illegible] a subsequent shutdown. This requirement is reasonable, based on engineerin[illegible] judgment, in view of other administrative controls to ensure that the flow [illegible] are operable. To further ensure EFW system alignment, flow path operabilit[illegible] verified following extended outages to determine no misalignment of valves [illegible] occurred. This SR ensures that the flow path from the 'Q' CST to the steam generators is properly aligned.~~

~~The functional test, performed once every 18 months, will verify that the flow path to the steam generators is open and that water reaches the steam generators from the emergency feedwater system. The test is done during shutdown to avoid thermal cycle to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater.~~

~~The automatic actuation circuitry testing and calibration will be performed per Surveillance Specification 4.1, and will be sufficient to assure that this circuitry will perform its intended function when called upon.~~

3.7.9
3.7.10

4.10 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM SURVEILLANCE

A1

Applicability

Applies to the surveillance of the control room emergency ventilation and air conditioning systems.

Objective

To verify an acceptable level of efficiency and operability of the control room emergency ventilation and air conditioning systems.

Specification

4.10.1 Each train of control room emergency air conditioning shall be demonstrated Operable:

LA1 BASES

SR 3.7.10.1

a. At least once per 31 days ~~on a staggered test basis~~ by:

1. Starting each unit and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤84°F D.B.

SR 3.7.10.2

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ±10%.

4.10.2 Each Control Room Emergency Ventilation System shall be demonstrated Operable:

LA1 Bases

SR 3.7.9.1

a. At least once per 31 days ~~on a Staggered Test Basis by initiating, from the Control Room, flow through the HEPA filters and charcoal adsorbers and~~ verifying that the system operates for at least 15 minutes.

SR 3.7.9.2 + <LATER> (5.0)

LATER

b. At least once per 18 months or 1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or 2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

   a. ≤2.5% for 2 inch charcoal adsorber beds, or

   b. ≤0.5% for 4 inch charcoal adsorber beds.

3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

3.7.9
3.7.10

SR 3.7.9.2
+<LATER>
(5.0)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

1. ≤ 2.5% for 2 inch charcoal adsorber beds, or

2. ≤ 0.5% for 4 inch charcoal adsorber beds.

+LATER

SR 3.7.9.2
SR 3.7.9.3
+<LATER (5.0)>

d. At least once per 18 months by:

+LATER

SR 3.7.9.2
+<LATER (5.0)>

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches of water while operating at a flowrate of 2000 cfm ±10%.

L16

SR 3.7.9.3

2. Verifying that on a Control Room high radiation test signal, the system automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

LA1
Bases

SR 3.7.9.2
+ <LATER>
(5.0)

e. After each complete or partial replacement of the HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

+LATER

Bases

A2

The purpose of the control room emergency ventilation system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with 100 percent capacity filter trains which consist of a prefilter, high efficiency particulate filters, charcoal adsorbers and a fan.

Since the emergency ventilation system is not normally operated, a periodic test is required to insure operability when needed. During this test the system will be inspected for such things as water, oil, or other foreign material; gasket deterioration, adhesive deterioration in the HEPA units; and unusual or excessive noise or vibration when the fan motor is running. Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

3.7.9
3.7.10

A2

Bases (Continued)

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52.

The operability of the control room emergency air conditioning Systems ensure that the ambient air temperature does not exceed the allowable temperature for the equipment and instrumentation cooled by this system and the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.

Operation of the systems for 15 minutes every month will demonstrate operability of the emergency ventilation and emergency air conditioning systems. All dampers and other mechanical and isolation systems will be shown to be operable.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

3.7.11

4.11 PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the penetration room ventilation system. A1

Objective

To verify an acceptable level of efficiency and operability of the penetration room ventilation system.

Specification

SR 3.7.11.2 & (LATER) (5.0)

4.11.1 At intervals not to exceed 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%). LATER

4.11.2 Initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.

SR 3.7.11.3 4.11.3 At intervals not to exceed 18 months, automatic initiation of the penetration room ventilation system shall be demonstrated.

SR 3.7.11.2 & (LATER) (5.0)

4.11.4a The tests and sample analysis of Specification 3.13.1a, b, & c. shall be performed at intervals not to exceed 18 months or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system. LATER

b. Cold DOP testing shall also be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing. L14

SR 3.7.11.1 4.11.5 Each circuit shall be operated at least 1 hour every month. This test shall be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal. LA3 TRM

3.7.11

A2

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration room leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of a sealed penetration room, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building pressure engineered safety features signal and initially requires no operator action.

Since the system is not normally operated, a periodic test is required to show that the system is available for its engineered safety features function. During this test the system will be inspected for such things as water, oil, or other foreign material, gasket deterioration in the HEPA units, and unusual or excessive noise or vibration when the fan motor is running.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per 18 months to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to Regulatory Guide 1.52.

Operation of the system each month for 1 hour will demonstrate operability of the active system components and the filter and adsorber system. If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

LAR A7

3.7.8

4.13 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To verify the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

A1

Specification

4.13.1 The emergency cooling pond shall be determined operable:

SR 3.7.8.1

1. At least once per 24 hours by verifying the pond's indicated water level is ≥ 5 feet.

SR 3.7.8.2 & Note

2. At least once per 24 hours during the period from June 1 through September 30 by verifying that the pond's average water temperature at the point of discharge from the pond is within its limit.

SR 3.7.8.3

3. At least once per 12 months by making soundings of the pond and verifying an average depth of 5 feet and that the contained water volume of the pond is within its limit. LA1 Bases

SR 3.7.8.3

4. At least once per 12 months by a visual inspection of the loose stone (riprap) placed on the banks of the pond and of the concrete slab spillway and verifying that the earth portions of the stone covered embankments and the spillway: LA1 Bases

   1. Have not been eroded or undercut by wave action, and

   2. Do not show apparent changes in visual appearance or other abnormal degradation from their as built condition.

Bases

The requirements of Specification 4.13 provide for verification of a sufficient water inventory in the emergency cooling pond to handle a DBA with a concurrent failure of the Dardanelle Reservoir. This specification ensures that Specification 3.11.1 is met. Monitoring temperature only during the period June 1 through September 30 of each year ensures that, during the hot summer months, the pond temperature limit is not exceeded. During other periods of the year the pond temperature will not have the potential to reach the temperature limit. Soundings are performed to ensure the water volume is within limits and that the indicated level is indicative of an equivalent water volume for accident mitigation. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions. Visual inspections are performed to ensure any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation sr be performed by a qualified engineer of any apparent changes in visual appearance or other abnormal degradation to determine operability.

A2

3.7

LA3

TRM

4.14 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Test for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferrer indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed startup source shall be leak tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4. The periodic leak test does not apply to the boronometer source. This source shall be tested for leakage at least once per 18 months.

<Add SR 3.7.12.1> M25

4.17 FUEL HANDLING AREA VENTILATION SYSTEM SURVEILLANCE

A1

Applicability

~~Applies to the surveillance of the fuel handling area ventilation system.~~

Objective

~~To verify an acceptable level of efficiency and operability of the fuel handling area ventilation system.~~

Specification

SR 3.7.12.2 & <LATER> (5.0)

LATER

4.17.1 ~~At intervals not to exceed 18 months, pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%).~~

4.17.2 ~~Initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.~~

4.17.3 a. ~~The tests and sample analysis of Specification 3.15.1.a, b, & c shall be performed within 720 system operating hours prior to irradiated fuel handling operations in the auxiliary building, and prior to irradiated fuel handling in the auxiliary building following significant painting, fire or chemical release in any ventilation zone communicating with the system.~~

b. ~~Cold DOP testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.~~

c. ~~Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.~~

L14

4.17.4 ~~The system shall be operated for at least 10 hours prior to initiation of irradiated fuel handling operations in the auxiliary building if it has not been operated for at least 10 hours within the previous 30 days.~~

A2

Bases

~~Since the fuel handling area ventilation system may be in operation when fuel is stored in the pool but not being handled, its operability must be verified before handling of irradiated fuel. Operation of the system for 10 hours before irradiated fuel handling operations and performance of Specification 4.17.3 will demonstrate operability of the active system components and the filter and adsorber systems.~~

3.7.12

A2

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop and air distribution should be determined once every 18 months to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to Regulatory Guide 1.52.

LAR A7

3.7.11

(LATER) (4.0)

assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks. [1]

LATER

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. [2]

5.2.3 Penetration Room Ventilation System

LA1 Bases

This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. [3]

REFERENCES

(LATER) (4.0)

(1) FSAR Section 5.1

LATER

(2) FSAR Section 5.2.5

LA1 Bases

(3) FSAR Section 6.5

3.7

6.12.5 Special Reports

<LATER>
(5.0)

~~Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.~~ LATER

a. ~~Deleted~~ A1

<LATER>
(3.3D)

b. ~~Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.~~ LATER

c. ~~Deleted~~ A1

<LATER>
(5.0)

d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18. LATER

e. ~~Miscellaneous Radioactive Materials Source Leakage Tests Specification 3.12.2.~~ L11

f. ~~Deleted~~

g. ~~Deleted~~

h. ~~Deleted~~

i. ~~Deleted~~ A1

<LATER>
(3.8)

j. ~~Degraded Auxiliary Electrical Systems, Specification 3.7.2.H.~~ LATER

<LATER>
(3.3D)

k. ~~Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1~~

l. ~~Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1~~ LATER

m. ~~Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.~~

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.7: Plant Systems

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

<u>3.7 L1</u>

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The number of main steam safety valves (MSSVs) required to be OPERABLE is reduced based on the number required to perform the safety function at specific power levels. The MSSVs are considered as potential event initiators through inadvertent opening and depressurization of the secondary system. Current Technical Specifications only require 14 MSSVs to be OPERABLE. The control of inoperable MSSVs will be the same as controls for the currently allowed inoperable MSSV. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. The MSSVs also provide overpressurization protection for decreased heat removal events. Requirements are included to reduce reactor power well within the time frame during which the MSSVs were previously allowed to be inoperable with no action. Reducing the high flux trip setpoint provides assurance that sufficient MSSV capacity is available to mitigate the effects of an overpressure event during operation with less than 14 MSSVs. A reduced power reactor trip will result in consequences within those of previously analyzed accidents. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are to be taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for MSSVs is based on the capability to prevent an overpressurization event. The methodology for determination of the number of MSSVs includes a reduced reactor power trip setpoint to limit the thermal energy required to be relieved. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L2

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The MSSVs are considered as potential event initiators through inadvertent opening and depressurization of the secondary system. The control of inoperable MSSVs will be the same as controls for the currently allowed inoperable MSSV. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. The main steam safety valves (MSSVs) are proposed to be allowed to be setpoint tested in MODE 3 during startup. Currently the MSSVs are required to have met the surveillance requirements, including setpoint testing, prior to heating the reactor above 280°F. The MSSVs will still be required to be OPERABLE with their setpoints properly adjusted (prior to heatup above 280°F). Plant experience with setpoint adjustment provides reasonable expectation that the MSSVs are capable of performing their safety function to prevent an overpressurization event. Also, performing this test at conditions closer to actual operating conditions minimizes any potential for inaccuracy due to differences between test conditions and operating conditions. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure compliance with the limiting condition for operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety for MSSVs is based on the capability to prevent an overpressurization event. The plant experience with MSSV setpoint adjustment is incorporated into procedures which provide assurance of proper adjustment, and if needed, confirmation of the setpoint early in the startup to maintain the capability of the MSSVs to perform their function. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L3

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The change in the Required Action does not result in any hardware changes. The change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed). The change provides consistency between the Required Actions and Applicable conditions for the LCO. Further, the change of Required Actions does not significantly increase the consequences of an accident because the change does not affect the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters, from that resulting from the original analysis.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for unit conditions during which analysis assumes the equipment to function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The Required Actions are revised to be consistent with the Applicability for the equipment. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L4

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any hardware changes. The closed main steam isolation valves (MSIVs) or main feedwater isolation valves (MFIVs) are not assumed to be an initiator of any analyzed event. The consequences of any event occurring with the valves already closed will not be significantly increased since the isolation valves are already in their required position, and the closure time is zero which is less than the assumed closure time if the valves were open. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change allows continued operation in these conditions since the valves have already performed their safety function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The pertinent margin of safety associated with the isolation valve closure is provided by the time associated with the closure of the valves following an event. Since inoperable valves will be closed and maintained closed, the required closure time will be met and the change does not result in a significant reduction in a margin of safety.

3.7 L5

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs) are used to support mitigation of the consequences of an accident; however, they are not considered the initiator of any previously analyzed accident. As such the proposed revision of the Surveillance Frequency will not significantly increase the probability of any accident previously evaluated. Since the function of the isolation valves continues to be verified on a periodic basis, and the valves continue to be required to be OPERABLE, the change of the Surveillance Frequency will not reduce the capability of required equipment to mitigate the event. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety associated with the MSIV and MFIV is provided by their closure capability following an event. Since testing will continue to confirm the required parameters for these valves, this change does not involve a significant reduction in a margin of safety.

3.7 L6

Not Used.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L7

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters, from that of the analyses considering the original Completion Time.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L8

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The current Technical Specifications require a shutdown if both emergency feedwater (EFW) pumps or their associated flow paths are inoperable, and the nonsafety related auxiliary feedwater (AFW) pump is available. However, the AFW pump is not required to be tested or verified to be available on any periodic basis. A change is proposed to not require the shutdown depending on the nonsafety related equipment, but rather leave this option to the licensee based on current knowledge of plant equipment and capability. Inoperable EFW equipment is not considered as an initiator of any previously evaluated accident. Therefore, the change does not increase the probability of an accident previously evaluated. Previously evaluated accidents do not depend on the nonsafety related AFW pump to mitigate consequences. However, as with any system, if it is available to mitigate an accident, it may be used. Therefore, the proposed change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Required Action to initiate restoration of reliable safety related equipment has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and the potential impact of failure of nonsafety related equipment. Therefore, the propose Required Action does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L9

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The emergency feedwater (EFW) pumps are used to support mitigation of the consequences of an accident; however, EFW is not considered as the initiator of any event. Therefore, the proposed revision will not increase the probability of any accident previously evaluated. Since the function of the EFW pumps continues to be verified on a periodic basis, and the pumps continue to be required to be OPERABLE, the change of the Surveillance Frequency will not reduce the capability of required equipment to mitigate the event. As discussed in NUREG-1366, Section 9.1, industry studies indicate that EFW pump testing on a monthly basis may be contributing to equipment unavailability and that changing the test Frequency to quarterly is reasonably expected to increase the availability of the EFW system. Therefore, this change does not involve an increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety associated with the EFW pumps is provided by their flow capability following an event. Since testing will continue to confirm the required parameters for these pumps, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L10

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The emergency feedwater (EFW) pumps are used to support mitigation of the consequences of an accident; however, EFW is not considered as the initiator of any event. Therefore, the proposed revision will not significantly increase the probability of any accident previously evaluated. Since the function of the EFW pumps continues to be verified on a periodic basis, and the pumps continue to be required to be OPERABLE, the change of the Surveillance Frequency will not reduce the capability of required equipment to mitigate the event. This change also excludes requirements to perform functional testing of the motor driven EFW pump and it's associated train during MODE 4 when any steam generator is relied upon for heat removal. This presents requirements which are consistent with those proposed for the actuation system. During operation in this MODE, the time period for response to an event which requires emergency feedwater initiation is sufficient to allow for operator action. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety associated with the EFW system is provided by its capability to provide flow to the steam generators following an event. Since testing will continue to confirm the required parameters for the system, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L11

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any changes in hardware or methods of operation. The change in the submittal of "after the fact" information is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the administrative requirements for submittal of information and do not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L12

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The proposed change in the conditions of the Frequency for the performance of a Surveillance Requirement does not result in any hardware changes. Neither the EFW system flow path verification, nor the EFW system flowpath configuration are considered as the initiator of any previously evaluated accident. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. The Surveillance will continue to provide timely recognition of EFW system impairment thus providing the operator an opportunity to provide system restoration. Further, the Surveillance will continue to be performed prior to operation that would result in sufficient core heat production that would require operation of the EFW System during a subsequent shutdown. Therefore, the proposed change to the conditions of the SR Frequency does not significantly increase the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Therefore, this change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L13

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

An extension of the Completion Time for a Required Action does not result in any hardware changes. The service water system is not considered as the initiator of a previously evaluated accidents. Therefore, this change does not significantly increase the probability of occurrence for initiation of any analyzed event. Further, neither the reason for the inoperability nor the Completion Time for performance of Required Actions significantly increases the consequences of an accident because the change does not change the assumed response of the equipment in performing its specified mitigation functions.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, neither the reason for the inoperability nor the short extension of the Completion Time interval involves a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L14

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The proposed change in the Surveillance Requirement does not result in any hardware changes. The ventilation systems are not considered as the initiator of any previously evaluated accidents. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. The ventilation systems are considered in the mitigation of consequences of some accidents. However, the length of time for operation of the system during surveillances is still sufficient to verify proper functioning of the system. Therefore, the proposed change to the Applicability does not significantly increase the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper functioning of the system through surveillance. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change will still ensure proper functioning of the system through surveillance. Therefore, this change does not involve a significant reduction in the margin of safety.

3.7 L15

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The proposed change in the Applicability and Required Actions does not result in any hardware changes. The analyses of concern are for a misloaded fuel assembly and a dropped fuel assembly. The spent fuel pool boron concentration is not considered as the initiator of either of these previously evaluated accidents. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. However, the spent fuel pool boron concentration is considered as an initial condition in the analysis of consequences of these accidents. Therefore, the Applicability will continue to include those conditions during which there is potential for these accidents, and the proposed Required Actions will initiate action to remove this potential. Therefore, the proposed change to the Applicability does not significantly increase the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L16

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The phrase "actual or simulated" in reference to the automatic initiation signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the function of the system functional test remains unaffected the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

Use of an actual signal instead of the existing requirement which limits use to a simulated signal, will not affect the performance of the surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between "actual" or "simulated." Therefore, the change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L17

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change in the Applicability and Required Actions does not result in any hardware changes. The analyses of concern are for a misloaded fuel assembly and a dropped fuel assembly. The penetration room ventilation system (PRVS) is not considered as the initiator of either of these previously evaluated accidents. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. Also, the PRVS is not considered in the mitigation of consequences of these accidents. Therefore, the proposed change to the Applicability does not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not involve a significant reduction in the margin of safety.

3.7 L18

Not Used.

3.7 L18

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The proposed change in the Required Actions for inoperable automatic actuation capability does not result in any hardware changes. The control room emergency ventilation system (CREVS) is not considered as the initiator of any previously evaluated accident. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. Also, under the revised ACTIONS, the CREVS will continue to be capable of providing its function in the mitigation of consequences of these accidents. Therefore, the proposed change to the ACTIONS does not significantly increase the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Therefore, this change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>3.7 L19</u>

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change will introduce the option to lock, seal, or otherwise secure the engineered safeguards (ES) valves for the service water system when OPERABILITY is required. Before this change, the only option was to lock the valves in the ES position. The method of verifying ES valve position is not an accident initiator and no hardware changes are proposed; therefore, the change does not significantly increase the probability of an accident. Expanding the methods available for verifying ES valve position does not significantly increase the consequences of a previously evaluated accident since the valves of interest are still placed in proper position for their safety function.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since expanding the methods of securing the ES valves in their actuated position has minimal impact on the availability of the systems. Furthermore, valve position surveillance, regardless of method of verification, is considered sufficient to provide system availability in the event of an accident.

ITS DISCUSSION OF DIFFERENCES

ITS Section 3.7: Plant Systems

1 NUREG-3.7.1 - The main steam safety valves (MSSV) Specification is reformatted to omit the table of specific lift setpoints and to replace the figure for determining the allowable power level and trip settings with predetermined values. The specific lift setpoints are currently required to be tested by current Technical Specification (CTS) Table 4.1-2, item 4. However, the CTS does not contain the specific setpoints. These setpoints are currently identified in the Inservice Testing (IST) Program and are adequately controlled therein under the design change and procedural control programs which include evaluations of changes in accordance with 10 CFR 50.59. Control of these setpoints is proposed to be retained in these programs. A minor editorial change is proposed to clarify that the 1% tolerance is only applicable to the "as-left" settings. The NUREG figure for determining the allowable nuclear overpower-high trip setpoint is provided for units which have MSSVs with different relief capacities. Since it would not be possible to predetermine which valves would be inoperable for the condition, a figure is not provided to calculate the required trip setpoint. However, all MSSVs at ANO-1 are of the same relieving capacity. Therefore, the allowable setpoint for the trip function can be predetermined based on the minimum number of OPERABLE valves per steam generator. This evaluation has been done and provided in a new Table 3.7.1-1, rather than by a figure, for the operators convenience. Also, the wording of Required Action A.1 is revised since the terminology of "reduced power requirement" from the figure is not used in the new Table. The proposed wording is consistent with the wording of Required Action A.2.

The LCO is revised to require that 14 MSSVs (7 on each main steam line) be OPERABLE regardless of power level. This means that Condition A merely allows continued operation rather than restoring compliance with the LCO. The NUREG-1430 Required Action A.1 restores compliance with the LCO and negates the requirement to change the setpoint in Required Action A.2 and control the setpoint during continued operation. This LCO change ensures that continued unit operation with an inoperable MSSV is in accordance with a Required Action.

Finally, Notes are added to the LCO and to SR 3.7.1.1 to retain the hydrotesting exception provided by CTS 3.4.1.2 Note *. This provides the capability to perform the hydrotesting using steam in lieu of water which would require additional supports due to the added weight. This exception is discussed ANO-1 license Amendment No. 90 (1CNA128405) and its associated request submittal (1CAN108401).

2 Incorporated TSTF-235, Rev. 1.

3 NUREG 3.7.2 & 3.7.3 - The Applicability of these LCOs is revised to MODES 1, 2, and 3, consistent with CTS 3.4.1. The MSIVs and MFIVs perform an accident mitigation function when there is significant mass and energy in the secondary system. In MODES 4, 5, and 6, the secondary side energy is low and these valves are not required to provide isolation. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

4 NUREG 3.7.2 & 3.7.3 – CTS 3.4.2 allowed action times for inoperable MSIVs and MFIVs are retained in the proposed ITS ACTIONS. Condition A entry conditions are expanded to include one or more MSIVs or MFIVs in both MODES 1 and 2. The unit design includes only two MSIVs and two MFIVs; therefore, closure of an MSIV or MFIV is not practical in either of these MODES. CTS 3.4.2 allows continued operation for 24 hours with either one or two inoperable MSIVs or MFIVs with no action required. This is retained in proposed Required Action A.1 for inoperable MSIVs or MFIVs. This proposed Completion Time allows time to prepare and implement activities necessary for restoration of OPERABILITY if the cause of the inoperability is restorable without a shutdown. Additionally, for MSIVs or MFIVs inoperable in MODE 3, the proposed Required Action C.1 is consistent with the CTS 3.4.2 Completion Time of 48 hours. Finally, the CTS 3.4.2 Completion Time of 24 hours to exit the MODE of Applicability is retained in Required Action D.1. Although the main steam and feedwater systems are not credited as a closed system, under normal conditions they do not provide a direct path from the reactor building atmosphere to the environment. Therefore, these Completion Times are reasonable, and provide for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown. This change is consistent with current license basis.

5 Not used.

These changes are consistent with the current license basis.

6 NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

The 10 CFR 50.36 Criterion satisfied by the ITS LCOs was modified to preserve consistency with the ANO-1 license basis. The NUREG Criterion specified were modified to be consistent with the analysis assumptions regarding equipment availability and operating condition (i.e., MODE).

7 NUREG 3.7.2 - Incorporates TSTF-209, Rev 1.

ITS DISCUSSION OF DIFFERENCES

8 NUREG 3.7.2 & 3.7.3 – Incorporated TSTF-289.

The specific required closure (isolation) time for the MSIVs and MFIVs is not incorporated. These values are not included in the CTS, have been adequately controlled in the Inservice Test Program, and are proposed to continue to be administratively controlled. This change is consistent with current license basis.

In accordance with unit design and operation, automatic closure capability is bypassed at ≤ 750 psig in the secondary system to avoid unintentional closure during normal shutdowns. Therefore, a Note is included in the automatic actuation surveillance (SR 3.7.2.2 and SR 3.7.3.2) to indicate that automatic isolation capability is not required when the secondary system pressure is ≤ 750 psig which is consistent with CTS 3.5.1.16. This change preserves current license basis requirements and accommodates unit specific design characteristics.

9 NUREG 3.7.3 - The CTS 3.4.1.5 requirements are applicable to only the main feedwater isolation valves (MFIVs). The main feedwater control valves and other associated valves are not currently required by the CTS. The ITS is proposed to retain these requirements for only the MFIVs. The capability of the other valves has been adequately controlled without specific Technical Specifications and are proposed to continue to be administratively controlled. This change is consistent with current license basis.

10 NUREG 3.7.14 - This LCO is not adopted. The CTS does not include such requirements and the administrative controls in place provide adequate assurance of sufficient fuel pool water level. This change is consistent with current license basis.

11 NUREG 3.7.4 - The ANO-1 safety analysis does not credit the atmospheric dump valves (ADVs) for events which meet the criteria of 10 CFR 50.36. Also, the CTS does not contain any requirements for the ADVs. Therefore, controls for these valves are proposed to continue to be administrative and not incorporated in the Technical Specifications. This change is consistent with current license basis.

12 NUREG 3.7.5 - The safety related emergency feedwater (EFW) system contains only two pumps and associated flow paths. All NUREG references to a third train or pump have been deleted. This change is consistent with current license basis.

13 NUREG 3.7.5 - Incorporates TSTF-101.

ITS DISCUSSION OF DIFFERENCES

14 NUREG 3.7.5 – Note 1 is omitted for SR 3.7.5.3 and SR 3.7.5.4. This testing is currently performed at low pressures to avoid either: a) making the system inoperable by tagging out the injection valves which would also open on the actuation signal, or b) injecting cold condensate into the steam generators. Valve and pump actuation can be demonstrated at low pressures, and along with full pressure, manual opening of the steam admission valves and pump flow testing, adequately demonstrates the capability of the system to perform these required safety functions. This change is consistent with current license basis.

Additionally, the wording of Note 2 for SR 3.7.5.3 and SR 3.7.5.4 revised for clarity and consistency with the Applicability. The "applicable" MODES are addressed only in the portion of the Specification entitled "APPLICABILITY" (with the exception of where applicable SRs of one specification are referenced by another specification, e.g., when a shutdown specification identifies the "applicable" SRs from the operating specification rather than repeat each "required" SR). Thus, Note 2 has been modified to clearly correlate with the Applicability. These changes are consistent with the NUREG Writer's Guide, and current license basis (CTS 3.4.3).

15 NUREG 3.7.5.5 – CTS 4.8.1.c requires this surveillance be performed only on manual valves. This is acceptable because it verifies the position of those valves that would not be easily detected through installed instrumentation and indication available to the operator or through the performance of a pump surveillance. In addition, this SR effectively replicates the requirements of SR 3.7.5.1 which must be performed prior to entry into the MODE of Applicability for this Specification. This change is consistent with current license basis.

In addition, the unit specific designation for the "Q" condensate storage tank (QCST) was provided to clarify which condensate storage tank is the subject of this SR (reference CTS 4.8.1.c). This change is consistent with current license basis.

16 NUREG 3.7.5 - The unit design does not include EFW pump suction pressure interlocks. Therefore, SR 3.7.5.6 and SR 3.7.5.7 are not incorporated. This change is consistent with current license basis.

17 NUREG 3.7.7 - The ANO-1 safety analysis does not credit the intermediate cooling water system for events which meet the criteria of 10 CFR 50.36. The safety related cooling water requirements are met by the service water system (see SAR Section 9.3). Therefore, only the service water system is proposed to be incorporated in the Technical Specifications. This change is consistent with current license basis.

18 NUREG 3.7.8 - The service water pumps are used in normal operation and, since they are already running, do not get an engineered safety actuation signal. The pumps will automatically restart following restoration of power subsequent to a bus undervoltage. Therefore, SR 3.7.8.3 is unnecessary and omitted. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

19 NUREG 3.7.9 - The ANO-1 ultimate heat sink does not utilize cooling towers, nor cooling tower fans. Therefore, the ACTIONS related to fans and SR 3.7.9.3 are not applicable. This change is consistent with current license basis.

20 NUREG 3.7.9 - SR 3.7.9.1, SR 3.7.9.2, and SR 3.7.9.3 are revised to verify the appropriate parameters for an emergency cooling pond consistent with CTS 4.13. ITS SR 3.7.8.3 will verify the pond contains the necessary volume when the water level is ≥ 5 ft, and ITS SR 3.7.8.1 will verify the pond level is ≥ 5 ft on a more frequent basis. The Frequency for ITS SR 3.7.8.3 is every 12 months since the degradation of the pond is gradual. ITS SR 3.7.8.2 is limited to only require the temperature verifications during the summer months when there is sufficient potential to exceed the limits to warrant the surveillance. These changes are consistent with current license basis.

21 NUREG 3.7.1 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.
B 3.7.1 BACKGROUND - Only 8 MSSVs are provided for each SG header.
B 3.7.1 ASA - Revised discussion of applicable transients and accidents in accordance with the current SAR.
B 3.7.1 LCO – Only 7 of the 8 MSSVs on each header are required for mitigation from full power.
B 3.7.1 LCO - The discussions of OPERABILITY are revised to prevent misinterpretation. These paragraphs incorrectly imply that they contain all requirements for OPERABILITY. They are revised to indicate that these are required to attain OPERABILITY but that compliance with these does not necessarily assure OPERABILITY. This is only determined by compliance with the definition of OPERABLE/OPERABILITY.
B 3.7.1 RA B.1 & B2 - The entry condition description is revised to match the Specification requirements.
B 3.7.1 References - A reference to Framatome Document 86-1266156-00, "ANO-1 Overpressure Protection," dated October 31, 1997, has been added to provide a reference for the MSSV relief capacity.

ITS DISCUSSION OF DIFFERENCES

22 NUREG 3.7.2 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.

B 3.7.2 BACKGROUND - Revised discussion of isolation signal to refer to more detailed description of initiating signals.

B 3.7.2 ASA - Revised discussion of Applicable Safety Analyses to be consistent with the unit specific analyses and license basis.

B 3.7.2 LCO - The discussions of OPERABILITY are revised to prevent misinterpretation and to be consistent with the unit specific analyses and license basis. These paragraphs incorrectly imply that they contain all requirements for OPERABILITY. They are revised to indicate that these are required to attain OPERABILITY but that compliance with these does not necessarily assure OPERABILITY. This is only determined by compliance with the definition of OPERABLE/OPERABILITY.

B 3.7.2 RA B.1 - The condition description is corrected for consistency with similar statements throughout the ITS Bases and with the wording of the Condition.

B 3.7.2 RA D.1 and D.2 - The condition description is corrected for accuracy.

23 NUREG 3.7.3 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.

B 3.7.3 BACKGROUND - Revised discussion of Emergency Feedwater Initiation and Control (EFIC) System to refer to more detailed description of initiating signals, and omit non-applicable discussions. Revised discussions of the purpose of MFIVs to be consistent with the unit specific analyses and license basis.

B 3.7.3 ASA - Revised discussion of Applicable Safety Analyses to be consistent with the unit specific analyses and license basis.

B 3.7.3 LCO - The discussions of OPERABILITY are revised to prevent misinterpretation. These paragraphs incorrectly imply that they contain all requirements for OPERABILITY. They are revised to indicate that these are required to attain OPERABILITY but that compliance with these does not necessarily assure OPERABILITY. This is only determined by compliance with the definition of OPERABLE/OPERABILITY.

B 3.7.3 LCO - Revised discussion to omit non-applicable discussions based on the unit specific analyses and license basis.

B 3.7.3 RA E.1 and E.2 - The condition description is corrected for accuracy.

24 NUREG 3.7.17 Bases - This change incorporates a thyroid dose conversion factor reference to the defined term of DOSE EQUIVALENT I-131 in Section 1.1, Definitions.

ITS DISCUSSION OF DIFFERENCES

25 NUREG 3.7.5 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.
B 3.7.5 General - The EFW system description is revised to reflect unit design and nomenclature.
B 3.7.5 General - Revised discussion of Emergency Feedwater Initiation and Control (EFIC) System to refer to more detailed description of initiating signals, and omit non-applicable discussions.
B 3.7.5 RA C.1 and C.2 - The condition description is corrected for accuracy.
B SR 3.7.5.1 - Clarification is provided for the "correct" position for automatic valves which may reposition upon an actuation signal.

26 NUREG 3.7.6 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.
B 3.7.6 BACKGROUND - The CST description is revised to reflect unit design.
B 3.7.6 ASA & LCO - The CST discussion of the applicable safety analysis is revised to be consistent with the unit specific analyses and license basis.
B 3.7.6 LCO - The discussion is clarified to identify the necessary volume if both units are relying on the "Q" CST, T-41B, and to revise the associated levels based on the latest calculations.
B 3.7.6 APPLICABILITY - The discussion is revised to address all conditions; "MODE with steam generators not being relied upon for heat removal" was missing.
B 3.7.6 RA B.1 & B.2 - The Required Actions do not provide a time for entry into MODE 4. However, the discussion of "an additional 6 hours" implies that MODE 4 must be entered within 12 hours. Since there is no such requirement, this misleading statement is omitted.

27 NUREG 3.7.8 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.
B 3.7.8 BACKGROUND, ASA & LCO - The service water system description is revised to reflect unit design and nomenclature.

28 NUREG 3.7.8 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.
B 3.7.8 BACKGROUND & ASA - The UHS description is revised to reflect unit design.

29 NUREG 3.7.10 - Required Actions C.2.1 and D.2 are omitted since they are not consistent with the Applicability of the Specification. Further, retention would be of no consequence since as soon as the concurrent action of "immediately suspend movement of irradiated fuel assemblies" is complete, the Specification will no longer be applicable and the CORE ALTERATIONS would no longer be controlled by this Specification. Finally, omission of these Required Actions is consistent with the "bracketed" identification of similar Required Actions in NUREG-1430 Specification 3.3.16.

ITS DISCUSSION OF DIFFERENCES

30 NUREG 3.7.15 - Incorporates TSTF-070, Rev. 1.

The word "spent" was added to the revised Required Action A.2.2 to clearly establish that this applies to the spent fuel pool storage area consistent with the wording of Required Actions A.1 and A.2.1. This editorial change is consistent with the terminology used in the current license basis.

31 NUREG 3.7.10 - NUREG SR 3.7.10.4 is not adopted. Per Standard Review Plan Section 6.4, only control room emergency ventilation system designs with < 0.25 volume changes per hour are required to provide periodic verification of the pressurization capability for the control room. As indicated in SAR 9.7.2.1, the ANO-1 CREVS is based on ≥ 3 volume changes per hour. Therefore, this Surveillance is not adopted. This change is consistent with current license basis.

32 NUREG 3.7.10 - NUREG SR 3.7.10.5 is not incorporated into the proposed ITS. Although outside air is provided as makeup when the control room is isolated, the capacity of the makeup is administratively controlled to comply with the habitability analysis assumptions. This control has been adequate in the past and is proposed to continue as such. This change is consistent with current license basis. [Note, NUREG-1430 provides no Bases information for this Surveillance.]

33 NUREG 3.7.12 & 3.7.13 - NUREG SR 3.7.12.5 is not incorporated for the penetration room ventilation system since no such action (opening) of the damper is provided in the system. NUREG SR 3.7.13.5 is not incorporated for the fuel handling area ventilation system since no such dampers are provided in the system. These changes are consistent with current license basis.

34 NUREG 3.7.13 - The Applicability and Required Actions of the requirements for the Fuel Handling Area Ventilation System are revised to include only those requirements associated with the handling of irradiated fuel assemblies in the fuel handling area. This is consistent with CTS 3.15.1 and with the safety analysis assumptions for operation of the filtration system. This change is consistent with current license basis.

Included with this change is an ACTIONS Note to indicate that LCO 3.0.3 is not applicable (consistent with CTS 3.15.2). Since the movement of irradiated fuel could occur in the fuel handling area during operation in MODES 1, 2, 3, or 4, if the applicable Required Actions could not be met, LCO 3.0.3 would require shutdown. However, this is inappropriate since operation of the unit is unrelated to fuel movement in the fuel handling area. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

35 NUREG 3.7.13 - The LCO and Actions are revised to require the fuel handling area ventilation system to be in operation when moving irradiated fuel in the fuel handling area consistent with CTS 3.15.1 requirements. ITS SR 3.7.12.1 is also included to periodically verify the system to be in operation during fuel handling in the area. NUREG SR 3.7.13.1 and SR 3.7.13.3 are not incorporated for the fuel handling area ventilation system since the system is placed in service prior to irradiated fuel movement in the fuel handling area and is not started on an actuation signal. These changes are consistent with current license basis.

36 NUREG 3.7.12 & 3.7.13 - NUREG SR 3.7.12.4 and SR 3.7.13.4 are not adopted. Per Standard Review Plan Section 6.4, only control room emergency ventilation system designs with < 0.25 volume changes per hour are required to provide periodic verification of the pressurization capability for the control room. The penetration room ventilation system (PRVS) also provides ≥ 0.25 volume changes per hour. Therefore, this Surveillance is also not adopted for the PRVS. The fuel handling area ventilation system (FHAVS) is not designed to pressurize the fuel handling area. Rather it provides a suction from the area immediately above the fuel pool. Therefore, the pressurization test is also not adopted for the FHAVS. These changes are consistent with current license basis.

37 NUREG 3.7.10 - The Note in Required Action C.1 is not required for this unit since the toxic gas mode of operation is the same as the radiation protection (emergency) mode, i.e., isolation, filtration, and pressurization with makeup air. The wording of Required Action C.1 is also revised to reflect that the CREVS must be placed in operation since there is only the emergency mode of operation, i.e., CREVS does not operate in a "normal" operation mode. This change is consistent with current license basis.

38 NUREG 3.7.18 - The unit safety analysis does consider a steam generator inventory; however, the inventory assumed in the analysis for a main steam line break is conservatively considered to be well above the level at which the steam generator aspirator ports would be flooded. Administrative controls have been sufficient to assure compliance with the safety analysis assumption, and an upper steam generator level is not included in the CTS. Therefore, the controls for these valves are proposed to continue to be administrative and not incorporated in the ITS. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

39 NUREG 3.7.10 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.

B 3.7.10 Background - The CREVS description is revised to reflect unit design.

B 3.7.10 ASA - The CREVS discussion of the applicable safety analysis is revised to be consistent with the unit specific analyses and license basis.

B 3.7.10 LCO - The discussion is revised to identify the correct components, i.e., no heater, demister or valves, and to use unit specific terminology.

B 3.7.10 Condition C - The discussion is revised to omit a misleading statement. Placing the system in operation does not ensure that "the remaining train is OPERABLE."

B 3.7.10 SR 3.7.10.1 - The discussion is revised to reflect unit design, i.e., without heaters.

B 3.7.10 SR 3.7.10.2 - The statement regarding compliance with Regulatory Guide 1.52 is redundant to the requirements of the VFTP and is unnecessary.

40 NUREG 3.7.11 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.

B 3.7.11 Background - The CREACS description is revised to reflect unit design.

B 3.7.11 ASA, LCO and Applicability - The CREACS discussion of the applicable safety analysis is revised to also address the habitability requirements portion of the license basis.

B 3.7.11 Condition B and Condition C - The condition description is corrected for accuracy.

B 3.7.11 Condition C - The discussion is revised to omit a misleading statement. Placing the system in operation does not ensure that "the remaining train is OPERABLE," and the system does not automatically actuate.

41 NUREG 3.7.12 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.

B 3.7.12 Background, ASA, LCO, and Applicability - The PRVS description is revised to reflect unit design.

B 3.7.12 Required Action A.1 - The condition description is corrected to identify that the PRVS supports mitigation of reactor building leakage, not support the ECCS.

B 3.7.12 Required Actions B.1 and B.2 - The condition description is corrected for accuracy.

B 3.7.12 SR 3.7.12.1 - The discussion is revised to reflect unit design, i.e., without heaters.

B 3.7.12 SR 3.7.12.2 - The statement regarding compliance with Regulatory Guide 1.52 is redundant to the requirements of the VFTP and is unnecessary.

ITS DISCUSSION OF DIFFERENCES

42 NUREG 3.7.13 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.

B 3.7.13 BACKGROUND, ASA, and LCO - The FHAVS description is revised to reflect unit design.

B 3.7.13 SR 3.7.13.2 - The statement regarding compliance with Regulatory Guide 1.52 is redundant to the requirements of the VFTP and is unnecessary.

43 NUREG 3.7.6 and Bases - Incorporates TSTF-140, except for the incorporation of the criterion in the Applicable Safety Analyses, as was described in DOD-6.

44 NUREG 3.7.15 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures.

B 3.7.15 All - The Spent Fuel Pool Boron Concentration Bases discussions are revised to reflect unit specific design and analysis.

B 3.7.17 Background, ASA, and LCO - The secondary specific activity Bases discussions are revised to reflect unit specific design and analysis. This change also incorporates TSTF-173.

45 NUREG 3.7.16 Bases - Incorporates TSTF-210.

46 ITS SR 3.7.5.6 – This change incorporates CTS 4.8.1.e.4 requirements to verify that feedwater is delivered to each steam generator using the electric motor-driven EFW pump. This SR is required to be performed on an 18 month Frequency as established in CTS 4.8.1.e. The addition of this SR complements NUREG SR 3.7.5.5 in verifying that feedwater can actually be delivered to the steam generators. This change is consistent with current license basis.

47 NUREG SR 3.7.5.5 and Bases – Incorporates TSTF-268.

48 NUREG 3.7.6 and Bases - Incorporates TSTF-174.

ITS DISCUSSION OF DIFFERENCES

49 NUREG-3.7.6 and Bases (ITS 3.7.14 and Bases) - Incorporates TSTF-255, Rev 1.

50 NUREG 3.7.12 and Bases (ITS 3.7.11 and Bases) Condition B has been revised to also apply when both PRVS trains are inoperable. CTS 3.13.1 requires two independent circuits of the PRVS to be operable. If one circuit of PRVS is made or found to be inoperable for any reason, 3.13.2 allows operation during the succeeding seven days provided the other circuit is operable. Failure to meet the requirements of 3.13.1 or 3.13.2 results in performing the actions of 3.13.3, which requires placing the reactor in the cold shutdown condition within 36 hours. NUREG 3.7.12 does not contain a Condition for both trains inoperable. Therefore, LCO 3.0.3 would be invoked. The CTS for PRVS does not require entry into LCO 3.0.3 since actions are provided in CTS 3.13.3, which would result in placing the reactor in cold shutdown in 36 hours, similar to the shutdown requirements of LCO 3.0.3. This change is consistent with the current license basis.

51 NUREG 3.7.1 Bases and NUREG 3.7.17 Bases - The term "AOO" is used in the GDCs, but the ANO-1 license basis is contingent upon discussion of "abnormalities" as defined and listed in SAR, Section 14.1. The ANO-1 SAR was written partially based on the guidance given in a "Guide to the Organization and Contents of Safety Analysis Reports" issued by the Atomic Energy Commission on June 30, 1966. This document discusses what transients or "abnormalities" should be considered for Core and Coolant Boundary Protection Analysis. Statements concerning the GDC criteria are modified in the ITS to reference the current license basis description in the Unit 1 SAR.

52 NUREG SR 3.7.11.1 and Bases - NUREG SR 3.7.11.1 has been deleted. The ITS will retain the current testing requirements specified in CTS 4.10.1.a and CTS 4.10.1.b. These surveillances were approved by the NRC for ANO-1 in a Safety Evaluation associated with Amendment 196 dated May 19, 1999. The ANO CREACS trains are not instrumented to an extent that would allow the specific requirement of NUREG SR 3.7.11.1 to be adequately performed. Generic Letter 89-13, Enclosure 2, describes a program acceptable to the NRC for heat exchanger testing. Frequent regular maintenance of a heat exchanger in lieu of testing for degraded performance is provided as an acceptable alternative action acceptable to the NRC. Periodic maintenance was credited for the CREACS in the ANO response to GL 89-13. The current combination of monthly functional testing and 18 month flow verification, when combined with preventative maintenance activities is sufficient to ensure the availability of the CREACS. This change retains the current license basis.

53 NUREG 3.7.16 Bases - This change provides unit specific revisions to discussions of design, analysis, or operational parameters or procedures.

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Seven ~~The~~ MSSVs shall be OPERABLE on each main steam line. ~~as specified in Table 3.7.1-1 and Figure 3.7.1-1~~ 3.4.1.2 1 Note *

⟨INSERT 3.7-1A⟩

APPLICABILITY: MODES 1, 2, and 3. 3.4.1

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each MSSV. NA

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One or more required MSSVs inoperable.	A.1 Reduce power in accordance with ~~to less than the reduced power requirement of~~ ~~Figure~~ Table 3.7.1-1. AND A.2 Reduce the nuclear overpower trip setpoint in accordance with ~~Figure~~ Table 3.7.1-1.	4 hours 36 ~~12~~ hours	3.4.2 1 2 NA 1
B. Required Action and associated Completion Time not met. OR One or more steam generators with less than (two) MSSVs OPERABLE.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours	3.4.2 3.4.2

<INSERT 3.7-1A>

------------------------NOTE--
During main steam system hydrotesting in MODE 3, one MSSV is required to be OPERABLE on each main steam line with lift setpoints adjusted to allow testing.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.7.1.1	-------------------NOTE------------------- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met during main steam system hydrotesting in MODE 3. --		NA 1 3.4.1.2 Note * 1
	Verify each required MSSV lift setpoint ~~per Table 3.7.1-1~~ in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program	T4.1-2 #4 edit

Combine into SR 3.7.1.1

Table 3.7.1-1 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER	LIFT SETTING (psig ± [3]%)
[2] MSSVs/steam generator	[1050]
[7] MSSVs/steam generator	[≤ 1100]

1

NA

Allowable Power Level and RPS Nuclear Overpower Trip Allowable Value Versus OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs OPERABLE (PER SG)	MAXIMUM ALLOWABLE POWER LEVEL (% RTP)	RPS NUCLEAR OVERPOWER TRIP ALLOWABLE VALUE (% RTP)
6	85.7	89.9
5	71.4	74.9
4	57.1	59.9
3	42.8	44.9
2	28.5	29.9
1	14.2	14.9

1

NA

Move to Bases (RA A.1 & A.2)

1

NA

$$\frac{WY}{Z} = SP; \quad RP = \frac{Y}{Z} \times 100\%$$

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1.

Y = Total OPERABLE MSSV relieving capacity per steam generator based on summation of individual OPERABLE MSSV relief capacities per steam generator [lb/hour].

Z = Required relieving capacity per steam generator of [6,585,600] lb/hour.

SP = Nuclear overpower trip setpoint (not to exceed W).

RP = Reduced power requirement (not to exceed RTP).

These equations are graphically represented below.
Operation is restricted to the area below and to the right of line BCDE.

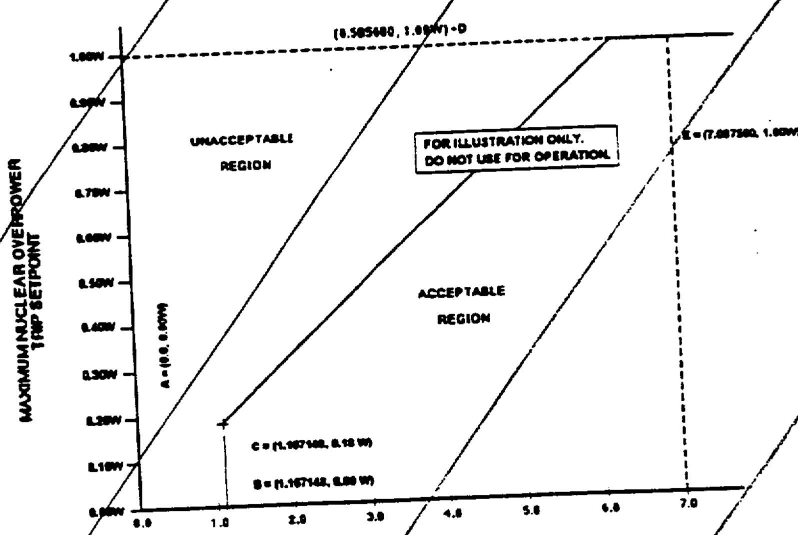


Figure 3.7.1-1 (page 1 of 1)
Reduced Power and Nuclear Overpower Trip Setpoint versus OPERABLE Main Steam Safety Valves

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE. 3.4.1.5

APPLICABILITY: MODE 1, 2, and 3. ~~MODES 2 and 3 except when all MSIVs are closed and [deactivated].~~ 3 3.4.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~MSIV~~ or more MSIV(s) inoperable in MODE 1 or 2.	A.1 Restore MSIV(s) to OPERABLE status.	24 ~~[8]~~ hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE ~~2~~ 3.	12 ~~6~~ hours
C. ---------NOTE--------- Separate Condition entry is allowed for each MSIV. ---------------------- One or more MSIV(s) inoperable in MODE ~~2~~ ~~or~~ 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	48 ~~[8]~~ hours Once per 7 days
D. Required Action and associated Completion Time of Condition ~~B~~ ~~or~~ C not met.	D.1 Be in MODE ~~3~~ 4. ~~AND~~ ~~D.2 Be in MODE 4.~~	24 ~~6~~ hours ~~12 hours~~

4 edit 3.4.2
4 3.4.2
NA 3.4.2 4
NA edit
4
3.4.2 4 7

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.2.1	--------NOTE-------- Only required to be performed in MODES 1 and 2. ------------------------ Verify ~~closure~~ isolation time of each MSIV is ~~≤ [6] seconds on an actual or simulated actuation signal.~~ within the limits specified in the Inservice Testing Program.	In accordance with the Inservice Testing Program ~~or [18] months~~	NA T4.1-2 #13 8
SR 3.7.2.2	------ NOTES ------ 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is <750 psig. ------------------------ Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months	NA 3.5.1.16 NA

9

3.7 PLANT SYSTEMS

3.7.3 ~~[~~Main Feedwater Isolation ~~Stop~~ Valves (MFIVs) ~~, Main Feedwater Control Valves (MFCVs), and Associated Startup Feedwater Control Valves (SFCVs)]~~ — 9

LCO 3.7.3 Two MFIVs ~~[Two] [MFSVs], [MFCVs], [or associated SFCVs]~~ shall be OPERABLE. — 3.4.1.5

APPLICABILITY: MODES 1, 2, and 3. ~~except when all [MFSVs], [MFCVs], [or associated SFCVs] are closed and [deactivated] [or isolated by a closed manual valve].~~ — 3.4.1, 3

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each ~~valve.~~ MFIV.
— NA, 9

CONDITION	REQUIRED ACTION	COMPLETION TIME
C ~~A~~. One ~~[MFSV] in one~~ or more ~~flow paths~~ MFIV(s) inoperable in MODE 3.	C ~~A~~.1 Close or isolate ~~[MFSV]~~ MFIV.	48 ~~[8 or 72]~~ hours — 4, 3.4.2, 9
	AND	
	C ~~A~~.2 Verify MFIV ~~[MFSV]~~ is closed or isolated.	Once per 7 days — NA
~~B. One [MFCV] in one or more flow paths inoperable.~~	~~B.1 Close or isolate [MFCV].~~	~~[8 or 72] hours~~ — 9, NA
	~~AND~~	
	~~B.2 Verify [MFCV] is closed or isolated.~~	~~Once per 7 days~~

(continued)

INSERT 3.7-7A, Conditions A & B — 4

<INSERT 3.7-7A>

				CTS
A. One or more MFIV(s) inoperable in MODE 1 or 2.	A.1	Restore MFIV(s) to OPERABLE status.	24 hours	3.4.2
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours	3.4.2

9

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C. One [SFCV] in one or more flow paths inoperable.~~	~~C.1 Close or isolate [SFCV].~~ ~~AND~~ ~~C.2 Verify [SFCV] is closed or isolated.~~	~~[8 or 72] hours~~ ~~Once per 7 days~~
~~D. Two valves in the same flow path inoperable for one or more flow paths.~~	~~D.1 Isolate affected flow path.~~	~~8 hours~~
D ~~E~~. Required Action and associated Completion Time of Condition C not met.	D ~~E~~.1 Be in MODE ~~3~~ 4. ~~AND~~ ~~E.2 Be in MODE 4.~~	24 ~~6~~ hours ~~12 hours~~

NA

9

NA

3.4.2

4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. -- Verify the isolation ~~closure~~ time of each MFIV ~~[MFCV], [MFCV], and [SFCV]~~ is ~~≤ [7] seconds on an actual or simulated actuation signal~~ within the limits provided in the Inservice Testing Program.	In accordance with the Inservice Testing Program ~~or [18] months~~

NA

9

T4.1-2 #14

8

<INSERT 3.7-8A>

<INSERT 3.7-8A>

SR 3.7.3.2	---NOTE--- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. ---		NA 3.5.1.16
	Verify that each MFIV actuates to the isolation position on an actual or simulated actuation signal.	18 months	NA

{ Reviewers Note: RSTS 3.7.17 has been renumbered and moved to fill ITS LCO 3.7.4 }

11

NA

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Vent Valves (AVVs)

LCO 3.7.4 [Two] AVVs [lines per steam generator] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required AVV [line] inoperable.	A.1	---------NOTE--------- LCO 3.0.4 is not applicable. --------------------- Restore required AVV [line] to OPERABLE status.	[7 days]
[B. Two or more required AVV [lines] inoperable.	B.1	Restore one AVV [line] to OPERABLE status.	24 hours]
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours 18 hours

11

NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each AVV.	[18] months
[SR 3.7.4.2	Verify one complete cycle of each AVV block valve.	[18] months]

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

12

LCO 3.7.5 Two ~~[Three]~~ EFW trains shall be OPERABLE. 3.4.3.1 3.4.3.2

-----NOTE-----
Only one EFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4 when steam generator is relied upon for heat removal. 3.4.3.1 edit

APPLICABILITY: MODES 1, 2, and 3, 3.4.3.1
MODE 4 when steam generator is relied upon for heat removal. 3.4.3.2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One steam supply to turbine driven EFW pump inoperable in MODE 1, 2, or 3.	A.1 Restore steam supply to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO	3.4.4.2 edit NA
B. One EFW train inoperable ~~[~~for reasons other than Condition A~~]~~ in MODE 1, 2, or 3.	B.1 Restore EFW train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO	3.4.4.3 NA

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
C. Required Action and associated Completion Time of Condition A ~~or B~~ not met. ~~OR~~ ~~Two EFW trains inoperable in MODE 1, 2, or 3.~~	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours ~~6~~ 18 hours	3.4.4.2 3.4.4.3 3.4.4.2 3.4.4.3 12
D. ~~Three~~ Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 --------NOTE--------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. --------------------- Initiate action to restore one EFW train to OPERABLE status.	Immediately	12 3.4.4.5 3.4.4.5
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately	3.4.4.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump~~s~~, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	4.8.1.b
SR 3.7.5.2	-------------------NOTE------------------- Not required to be performed for the turbine driven ~~EFW pump~~~~s~~, until ~~[~~24~~]~~ hours after reaching ~~[800]~~ ≥ 750 psig in the steam generators. -- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program ~~[31] days on a STAGGERED TEST BASIS~~	3.4.3.2 Note ** edit 4.8.1.a.1 4.8.1.a.5 13 4.8.1.a.2
SR 3.7.5.3	-------------------NOTE~~S~~------------------- ~~1. Not required to be performed until [24] hours after reaching [800] psig in the steam generators.~~ ~~2.~~ Not ~~applicable~~ required to be met in MODE 4 when steam generator is relied upon for heat removal -- Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[18]~~ 9 months	NA 4.8.1.e.2 NA edit 14 4.8.1.e.1 4.8.1.e.2 4.8.1.e.5

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.7.5.4	------------------NOTE~~S~~------------------ ~~1. Not required to be performed until [24] hours after reaching [800] psig in the steam generators.~~ 2. Not ~~applicable~~ required to be met in MODE 4 when steam generator is relied upon for heat removal --- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	~~[18]~~ months	NA & 4.8.1.e.2 (14) edit NA 4.8.1.e.2 4.8.1.e.3
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying ~~flow~~ valve manual alignment ~~flow~~ from the "Q" condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever ~~plant~~ unit has been in MODE 5 ~~or 6 for~~, MODE 6, or defueled for a cumulative period of > 30 days	4.8.1.c edit (15) (47)
~~SR 3.7.5.6~~	~~Perform a CHANNEL FUNCTIONAL TEST for the EFW pump suction pressure interlocks.~~	~~31 days~~	(16)
~~SR 3.7.5.7~~	~~Perform a CHANNEL CALIBRATION for the EFW pump suction pressure interlocks.~~	~~[18] months~~	(16)
SR 3.7.5.6	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	18 months	(46)

Generic term change
CST → QCST

edit

3.7 PLANT SYSTEMS

3.7.6 Q Condensate Storage Tank (CST)

LCO 3.7.6 The ~~[two]~~ CST ~~level(s)~~ shall be OPERABLE ~~≥ [250,000] gal~~. 43 3.4.1.3

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal. 3.4.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. The ~~[two]~~ CST ~~level(s) not within limits~~ inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter	43
	AND A.2 Restore CST ~~level(s)~~ to ~~within limit~~ OPERABLE status.	7 days	43 3.4.2
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	3.4.2
	AND B.2 Be in MODE 4 without reliance on steam generator for heat removal.	~~[~~18 hours~~]~~	3.4.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify CST ~~level~~ volume is ≥ ~~[250,000]~~ 32,300 gal. ✓lons	12 hours

edit NA

17

NA

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	--------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for emergency diesel generator made inoperable by CCW. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for decay heat removal made inoperable by CCW. --------------------- Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

17

NA

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	-------------------NOTE------------------- Isolation of CCW flow to individual components does not render CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	[18] months
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	[18] months

3.7 PLANT SYSTEMS

3.7.~~8~~ 7 Service Water System (SWS)

3.3.1 (C)
edit 3.3.1 (E)
3.3.1 (I)
3.3.4 (D)

LCO 3.7.~~8~~ 7 Two SWS ~~trains~~ loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. 3.3.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS ~~train~~ loop inoperable.	A.1 --------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for ~~emergency~~ diesel generator made inoperable by SWS. 2. Enter Applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for decay heat removal made inoperable by SWS. -------------------- Restore SWS ~~train~~ loop to OPERABLE status.	72 hours

edit
NA
edit
NA
edit 3.3.6

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
B. Required Action and associated Completion Time ~~of Condition A~~ not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours	3.3.6 edit 3.3.6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.7.1	-------------------NOTE------------------- Isolation of SWS flow to individual components does not render the SWS inoperable. -- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	NA NA
SR 3.7.7.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	9 ~~[18]~~ months	T4.1-2 #9 4.5.1.1.2(a)(2) 4.5.2.1.2.c.3
~~SR 3.7.8.3~~	~~Verify each SWS pump starts automatically on an actual or simulated actuation signal.~~	~~[18] months~~	18

ECP ~~UHS~~ 3.7.~~9~~ 8

3.7 PLANT SYSTEMS

3.7.~~9~~ 8 ~~Ultimate Heat Sink (UHS)~~ Emergency Cooling Pond (ECP) 3.11.1

LCO 3.7.~~9~~ 8 The ~~UHS~~ ECP shall be OPERABLE. 3.11.1

APPLICABILITY: MODES 1, 2, 3, and 4. 3.11.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
~~[A. One or more cooling towers with one cooling tower fan inoperable.~~	~~A.1~~	~~Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days]~~	19
A. ECP inoperable. ~~B. Required Action and associated Completion Time of Condition A not met~~ ~~OR~~ ~~UHS inoperable [for reasons other than Condition A].~~	A ~~B~~.1	Be in MODE 3.	6 hours	3.11.2
	AND A ~~B~~.2	Be in MODE 5.	36 hours	3.11.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~9~~ 8.1	Verify water level of ~~UHS~~ ECP is ≥ ~~[562]~~ 5 ft ~~[mean sea level]~~.	24 hours	20 4.13.1.1 3.11.1

(continued)

--- NOTE ---
Only required to be performed from June 1 through September 30.

ECP
~~UHS~~
3.7.~~9~~ 8

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.7.~~9~~ 8.2	Verify average water temperature ~~of UHS~~ at the point of discharge from the ECP is ≤ ~~[90]~~ 100 °F.	24 hours	20 4.13.1.2 3.11.1
~~SR 3.7.9.3~~	~~Operate each cooling tower fan for ≥ [15] minutes.~~	~~31 days~~	19
SR 3.7.8.3	Verify contained water volume of ECP ≥ 70 acre-ft at a water level of 5 ft.	12 months	20 4.13.1.3 4.13.1.4 3.11.1

3.7 PLANT SYSTEMS

3.7.~~10~~ 9 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.~~10~~ 9 Two CREVS trains shall be OPERABLE. 3.9.2.1 3.8.18

APPLICABILITY: MODES 1, 2, 3, and 4, ~~[5, and 6,]~~. 3.9.2.1 3.8.18
9 [During movement of irradiated fuel assemblies,]. 9
~~[During CORE ALTERATIONS].~~

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One CREVS train inoperable.	A.1	Restore CREVS train to OPERABLE status.	7 days	3.9.2.2
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1	Be in MODE 3.	6 hours	3.9.2.2
	AND			
	B.2	Be in MODE 5.	36 hours	3.9.2.2
9 [C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~.	C.1	~~-----NOTE----- Place in emergency mode if automatic transfer to emergency mode inoperable. ---------~~ Place OPERABLE CREVS train in emergency recirculation mode.	Immediately 2	37 NA NA 37
	OR			

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
[C. (continued)	~~C.2.1 Suspend Core ALTERATIONS.~~ ~~AND~~	~~Immediately~~	29 NA
	C.2.~~2~~ Suspend movement of irradiated fuel assemblies.	Immediately]	NA
[D. Two CREVS trains inoperable during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately	NA
	~~AND~~ ~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~]	29
E. Two CREVS trains inoperable during MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~10~~ 9.1	Operate each CREVS train for ~~[≥ 10 continuous hours with the heaters operating or (for system without heaters)~~ ≥ 15 minutes~~]~~.	31 days	4.10.2.a

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.7.~~10~~ 9.2	Perform required CREVS filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~	4.10.2
SR 3.7.~~10~~ 9.3	Verify each CREVS train actuates and ~~[or the~~ the control room isolates~~]~~ on an actual or simulated actuation signal.	~~[~~18~~]~~ months	4.10.2.d.2
~~SR 3.7.10.4~~	~~Verify one CREVS train can maintain a positive pressure of ≥ [0.125] inches water gauge relative to the adjacent [area] during the [pressurization] mode of operation at a flow rate of ≤ [3300] cfm.~~	~~[18] months on a STAGGERED TEST BASIS~~	31
~~SR 3.7.10.5~~	~~Verify the system makeup flow rate is ≥ [270] and ≤ [330] cfm when supplying the the control room with outside air.~~	~~[18] months~~	32

CREA~~T~~CS
3.7.~~[11]~~ 10

3.7 PLANT SYSTEMS

3.7.~~[11]~~ 10 Control Room Emergency Air ~~Temperature Control~~ Conditioning System (CREA~~T~~CS)

LCO 3.7.~~[11]~~ 10 Two CREA~~T~~CS trains shall be OPERABLE. 3.9.1.1

APPLICABILITY: MODES 1, 2, 3, and 4, ~~[5, and 6,]~~.
During movement of irradiated fuel assemblies~~,~~.
~~[During CORE ALTERATIONS]~~. 3.9.1.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One CREA~~T~~CS train inoperable.	A.1	Restore CREA~~T~~CS train to OPERABLE status.	30 days	3.9.1.2
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1	Be in MODE 3.	6 hours	3.9.1.2
	AND			
	B.2	Be in MODE 5.	36 hours	3.9.1.2
[C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~.	C.1	Place OPERABLE CREA~~T~~CS train in operation.	Immediately	NA
	OR			
	C.2	Suspend movement of irradiated fuel assemblies.	Immediately]	NA

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
[D. Two CREA~~T~~CS 9 trains inoperable during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~.	D.1	Suspend movement of irradiated fuel assemblies.	Immediately]	NA
E. Two CREA~~T~~CS 9 trains inoperable during MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.7.11.1~~	~~Verify each CREATCS train has the capability to remove the assumed heat load.~~	~~[18] months~~
SR 3.7.10.1	Verify each CREACS train starts, operates for at least 1 hour, and maintains control room air temperature ≤ 84°F.	31 days
SR 3.7.10.2	Verify system flow rate of 9900 cfm ±10%	18 months

52

3.7 PLANT SYSTEMS

3.7.~~12~~ 11 ~~Emergency~~ Penetration Room Ventilation System ~~(EVS)~~ (PRVS)

LCO 3.7.~~12~~ 11 Two PR ~~E~~VS trains shall be OPERABLE. 3.13.1

APPLICABILITY: MODES 1, 2, 3, and 4. 3.13.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One PR ~~E~~VS train inoperable.	A.1 Restore PR ~~E~~VS train to OPERABLE status.	7 days	3.13.2
B. Required Action and associated Completion Time not met. OR Both PRVS trains inoperable	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours	NA 3.13.3 50

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~12~~ 11.1	Operate each PR ~~E~~VS train for [~~≥ 10 continuous hours with the heaters operating or (for systems without heaters)~~ ≥ 15 minutes].	31 days	4.11.5
SR 3.7.~~12~~ 11.2	Perform required PR ~~E~~VS filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. 6	In accordance with the ~~[~~VFTP~~]~~ 6	3.13.1 4.11.1 4.11.2 4.11.4

(continued)

PR EVS
3.7.11

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.7.11.3	Verify each PR EVS train actuates on an actual or simulated actuation signal.	~~[18]~~ 6 months	4.11.3 3.13.1.f
~~SR 3.7.12.4~~	~~Verify one EVS train can maintain a pressure ≤ [ ] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of ≤ [3000] cfm.~~	~~[18] months on a STAGGERED TEST BASIS~~	36
~~SR 3.7.12.5~~	~~Verify each EVS filter cooling bypass damper can be opened.~~	~~[18] months~~	NA 33

FHAVS
~~FSPVS~~
3.7.~~13~~ 12

3.7 PLANT SYSTEMS

3.7.~~13~~ 12 Fuel ~~Storage Pool~~ Handling Area Ventilation System (F~~SP~~HAVS)

LCO 3.7.~~13~~ 12 ~~[Two] FSPVS trains~~ The FHAVS shall be OPERABLE and in operation. 35 3.15.1

APPLICABILITY: ~~[MODES 1, 2, 3, and 4,]~~ During movement of irradiated fuel assemblies in the fuel ~~building~~ handling area. 34 3.15.1

---NOTE---
LCO 3.0.3 is not applicable.
34 3.15.2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
~~A. One FSPVS train inoperable.~~	~~A.1~~	~~Restore FSPVS train to OPERABLE status.~~	~~7 days~~	NA
~~[B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.~~ ~~OR~~ ~~Two FSPVs trains inoperable in MODE 1, 2, 3, or 4.~~	~~B.1~~ ~~AND~~ ~~B.2~~	~~Be in MODE 3.~~ ~~Be in MODE 5.~~	~~6 hours~~ ~~36 hours]~~	NA 34
~~C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the fuel building.~~	~~C.1~~ ~~OR~~ ~~C.2~~	~~Place OPERABLE FSPVS train in operation.~~ ~~Suspend movement of irradiated fuel assemblies in the fuel building.~~	~~Immediately~~ ~~Immediately~~	NA

(continued)

FHAVS
~~FSPVS~~
3.7.~~13~~ 12

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. FHAVS ~~Two FSPVS trains~~ inoperable during movement of irradiated fuel assemblies in the fuel building or not in operation	A.1 Suspend movement of irradiated fuel assemblies in the fuel ~~building~~ handing area.	Immediately	3.15.2 35

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
~~SR 3.7.13.1~~	~~Operate each FSPVS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].~~	~~31 days~~	35
SR 3.7.~~13~~ 12.2	Perform required ~~FSPVS~~ FHAVS filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~	3.15.1 4.17.1 4.17.2 4.17.3
~~SR 3.7.13.3~~	~~Verify each FSPVS train actuates on an actual or simulated actuation signal.~~	~~[18] months~~	35
~~SR 3.7.13.4~~	~~Verify one FSPVS train can maintain a pressure ≤ [ ] inches water gauge with respect to atmospheric pressure during the [post accident] mode of operation at a flow rate ≤ [3000] cfm.~~	~~[18] months on a STAGGERED TEST BASIS~~	36
SR 3.7.12.1	Verify FHAVS in operation.	(continued) 12 hours	35 NA

FHAVS
~~FSPVS~~
3.7.~~13~~ 12

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.13.5~~	~~Verify each FSPVS filter bypass damper can be opened.~~	~~[18] months~~

33

10

3.7 PLANT SYSTEMS

3.7.14 Fuel Storage Pool Water Level

LCO 3.7.14 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 ---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.~~15~~ 13 Spent Fuel Pool Boron Concentration

LCO 3.7.~~15~~ 13 The spent fuel pool boron concentration shall be ≥ ~~[500]~~ 1600 ppm. 3.8.17

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool. 3.8.17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Spent fuel pool boron concentration not within limit.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------		NA
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately	NA
	AND		
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately	NA
	OR		
	A.2.2 Initiate action to perform ~~Verify by administrative means~~ a ~~[Region 2]~~ spent fuel pool verification ~~has been performed since the last movement of fuel assemblies in the spent fuel pool~~.	Immediately	30

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.~~15~~ 13.1	Verify the spent fuel pool boron concentration is ~~within limit.~~ ≥ 1600 ppm.	7 days

T4.1-3
#4
edit

49

3.7 PLANT SYSTEMS

3.7.~~16~~ 14 Spent Fuel ~~Assembly~~ Pool Storage

49

LCO 3.7.~~16~~ 14 The combination of initial enrichment and burnup of each ~~spent~~ fuel assembly stored in [Region 2] shall be within the acceptable ~~[burnup domain]~~ range of Figure 3.7.~~16~~ 14-1 or in accordance with Specification 4.3.1.1.

3.8.16 49

edit

APPLICABILITY: Whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.

3.8.16

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------NOTE--------- LCO 3.0.3 is not applicable. ---------------------- Initiate action to move the noncomplying fuel assembly from [Region 2].	Immediately

3.8.16

NA

SURVEILLANCE REQUIREMENTS

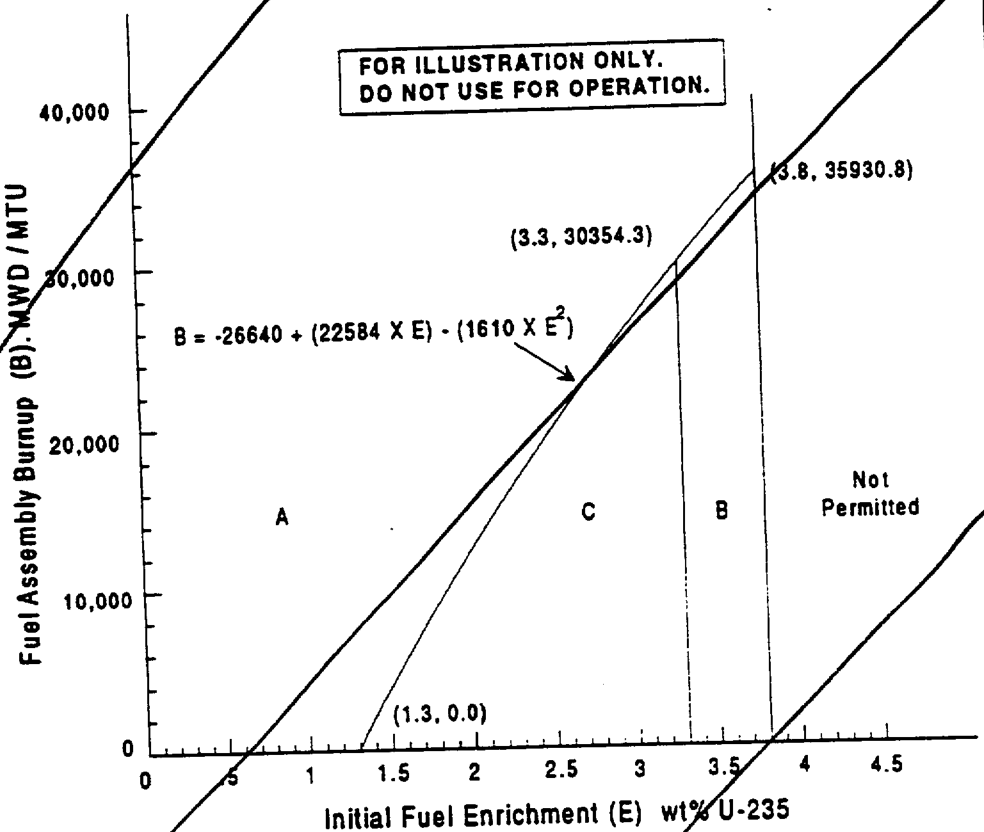
SURVEILLANCE		FREQUENCY
SR 3.7.~~16~~ 14.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.~~16~~ 14-1 or Specification 4.3.1.1.	Once ~~P~~prior to storing the fuel assembly in [Region 2]

edit
3.8.16
edit

<INSERT 3.7-37A>

Replace with ANO-1 CTS Figure 3.8.2

49



Category "A" Fuel - May be located anywhere within the storage racks.
Category "B" Fuel - Shall only be located adjacent to Category "A" Fuel or water holes within the storage racks.
Category "C" Fuel - Shall not be located adjacent to Category "B" Fuel.

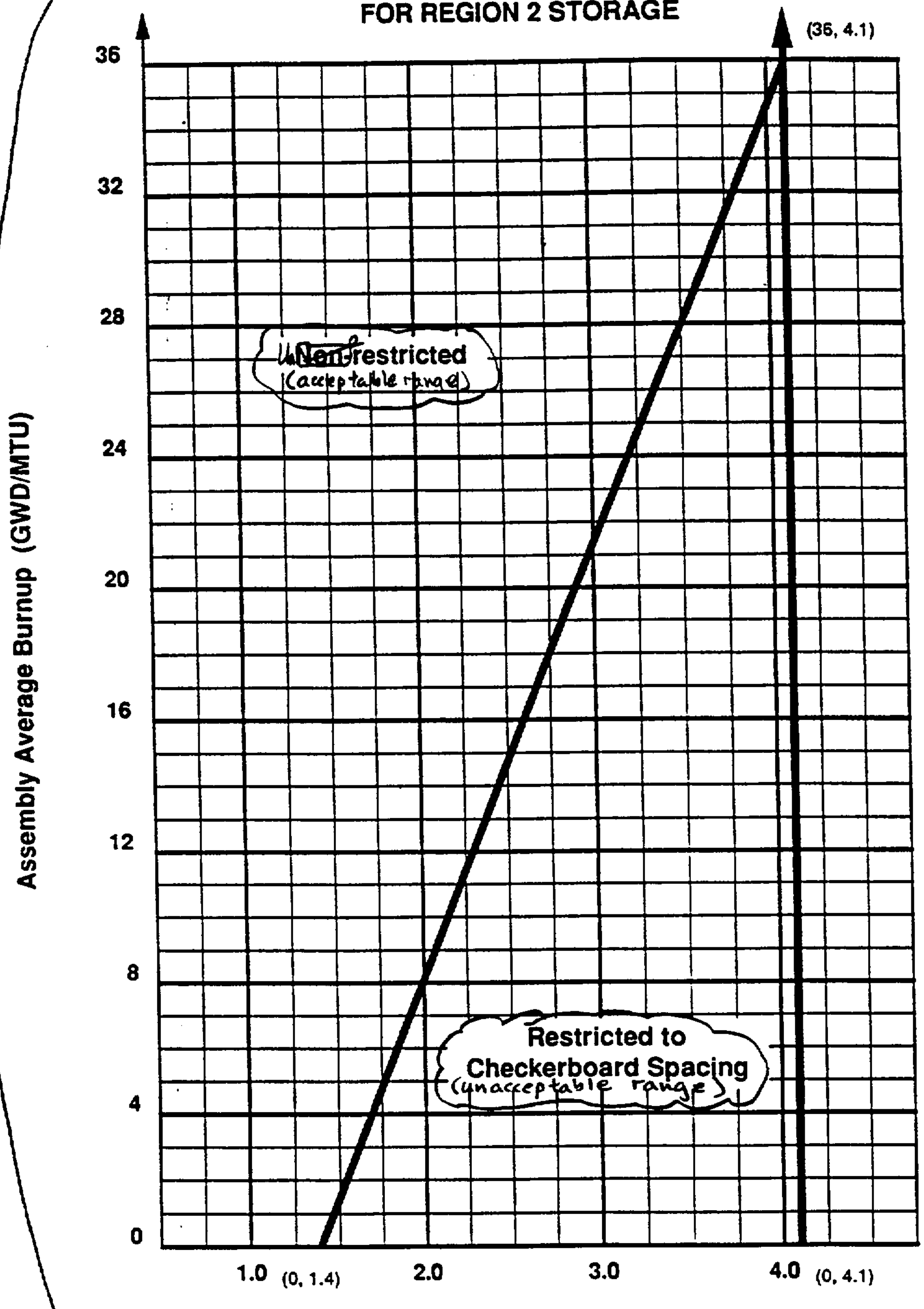
Figure 3.7.14-1 (page 1 of 1)
Burnup versus Enrichment Curve for
Spent Fuel Storage Racks

F3.8.2

INSERT 3.7-37A

~~FIGURE 3.8.2~~

MINIMUM BURNUP VS. INITIAL ENRICHMENT
FOR REGION 2 STORAGE



3.7 PLANT SYSTEMS

3.7.~~[17]~~ 4 Secondary Specific Activity

LCO 3.7.~~[17]~~ 4 The specific activity of the secondary coolant shall be ≤ ~~[0.10]~~ 0.17 μCi/gm DOSE EQUIVALENT I-131. 3.10

APPLICABILITY: MODES 1, 2, 3, and 4. T4.1-3,#7 T4.1-3,#10

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours	3.10
	AND		
	A.2 Be in MODE 5.	36 hours	3.10

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.7.~~[17]~~ 4.1 Verify the specific activity of the secondary coolant is ≤ ~~[0.10]~~ 0.17 μCi/gm DOSE EQUIVALENT I-131.	~~[31]~~ 31 days	T4.1-3 #5.b & Note 4

38

3.7 PLANT SYSTEMS

3.7.18 Steam Generator Level

LCO 3.7.18 Water level of each steam generator shall be less than or equal to the maximum water level shown in Figure 3.7.18-1.

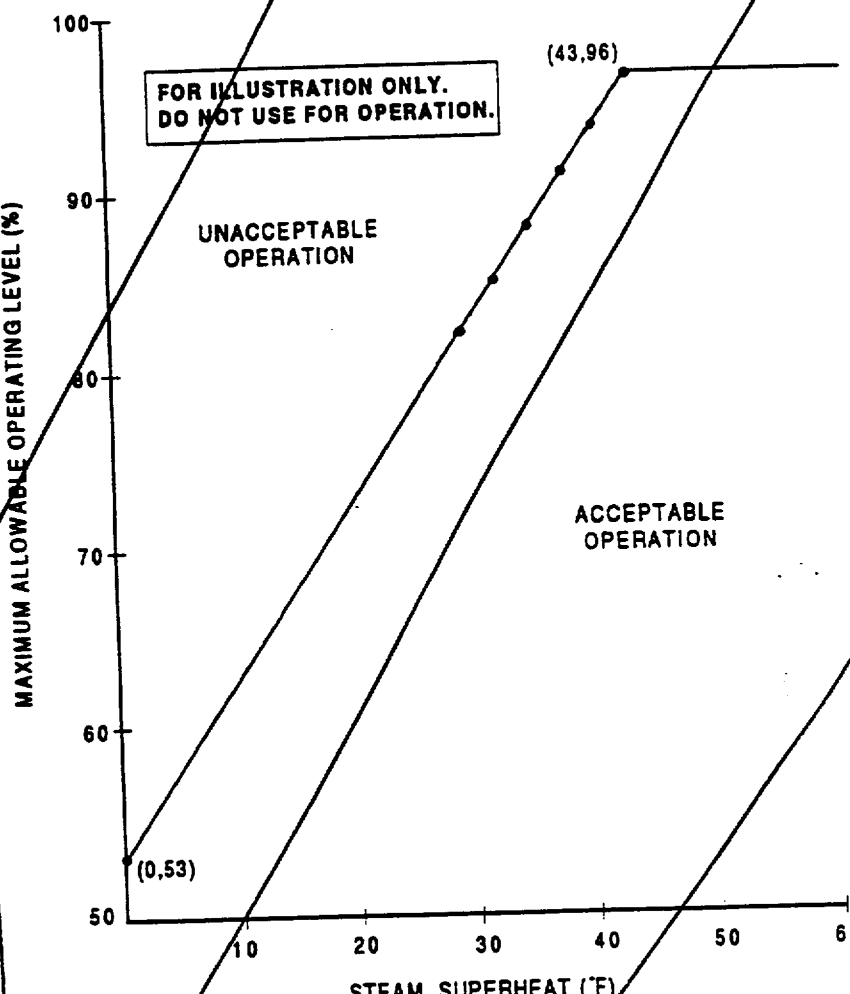
APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Water level in one or more steam generators greater than maximum water level in Figure 3.7.18-1.	A.1	Restore steam generator level to within limit.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify steam generator water level to be within limits.	12 hours



38

Figure 3.7.18-1 (page 1 of 1)
Maximum Allowable Steam Generator Level

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available. 21

the reactor building Eight 10.3

~~Nine~~ MSSVs are located on each main steam header, outside ~~containment~~, upstream of the main steam isolation valves, as described in the ~~F~~SAR, Section ~~[5.2]~~ (Ref. 1). The MSSV ~~rated~~ capacity ~~passes the full steam flow at [112% RTP with the valves full open]~~. ~~This meets~~ the requirements of the ASME Code, Section III ~~(Ref. 2)~~. The MSSV design includes staggered setpoints~~, according to Table 3.7.1-1 in the accompanying LCO,~~ so that only the needed number of valves will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open ~~all~~ the valves~~, following a turbine reactor trip~~. (Ref. 1) 1 edit

is adequate to meet

The total capacity of 14 MSSVs is greater than the total steam flow at 102% RTP.

edit

APPLICABLE SAFETY ANALYSES

The design basis of the MSSVs ~~comes from~~ Ref~~erence~~ 2) ~~and its purpose~~ is to limit secondary system pressure to ≤ 110% of design pressure when passing ~~100%~~ 102% of design steam flow~~.~~ ~~This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.~~ edit 51

(100% plus 2% heat balance error).

The MSSVs ensure that the design basis requirement is met for any abnormality or accident considered in the SAR.

may assume use

The events that ~~challenge the relieving capacity~~ of the MSSVs~~, and thus RCS pressure,~~ are those characterized as decreased heat removal events~~, and are presented in the [FSAR, Section [15.2] (Ref. 3). Of these, the full power turbine trip coincident with a loss of condenser heat sink is the limiting AOO. For this event, the Condenser Circulating Water System is lost and, therefore, the Turbine Bypass Valves are not available to relieve Main Steam System pressure. Similarly, MSSV relief capacity is utilized in the FSAR for]~~ mitigation of the following events: edit 21 edit

use may be assumed during

a. Loss of load ~~main feedwater;~~ (SAR, Chapter 14 (Ref. 3));

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~b. Steam line break;~~ 21

b ~~c~~. Steam generator tube rupture; and

Small break loss of coolant (Ref. 3).

c ~~d~~. ~~Excessive heat removal due to feedwater system malfunction.~~

<INSERT B3.7-2A>

In MODES 1 and 2, The MSSVs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 5). 6

<INSERT B3.7-2E>

LCO

The MSSVs ~~setpoints~~ are ~~established~~ provided to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires ~~[14]~~ fourteen MSSVs (seven on each main steam line) to be OPERABLE to ensure compliance with the ASME Code following DBAs initiated at full power. Operation with less than ~~a full~~ the required complement of MSSVs requires ~~s~~ limitation~~s~~ on unit THERMAL POWER and adjustment of the Reactor Protection System (RPS) nuclear overpower trip setpoint~~s~~. This effectively limits the Main Steam System steam flow while the MSSV relieving capacity is reduced due to valve inoperability. To be OPERABLE, lift setpoints must remain within limits, according to ~~Table 3.7.1-1 in the accompanying LCO~~ SR 3.7.1.1. 21 edit 1

<INSERT B3.7-2B>

The ~~OPERABILITY~~ safety function of the MSSVs is ~~defined as the ability~~ to open~~s~~ ~~within the setpoint tolerances~~, relieve steam generator overpressure, and reseat when pressure has been reduced. 21

~~The~~ OPERABILITY of the MSSVs ~~is determined by~~ requires periodic surveillance testing in accordance with the Inservice Testing Program.

<INSERT B3.7-2C>

The lift settings~~, according to Table 3.7.1-1 in the accompanying LCO,~~ correspond to ambient conditions of the valve at nominal operating temperature and pressure. 21

This LCO provides assurance that the MSSVs will perform the design safety function~~, to mitigate the consequences of accidents that could result in a challenge to the RCPB~~. edit 1

<INSERT B3.7-2D>

APPLICABILITY

~~In MODE 1 above [18]% RTP, the number of MSSVs per steam generator required to be OPERABLE must be within the acceptable region, according to Figure 3.7.1-1 in the~~ 1

(continued)

<INSERT B3.7-2A>

The full power turbine trip coincident with a loss of condensate heat sink establishes the required MSSV relief capacity (Ref. 4).

<INSERT B3.7-2B>

The minimum number of OPERABLE MSSVs per steam generator for various power levels and the associated maximum allowable nuclear overpower trip setpoint are identified in Table 3.7.1-1.

<INSERT B3.7-2C>

With all MSSVs OPERABLE, at least one MSSV per steam generator is set at 1050 psig nominal, while the remaining MSSVs per steam generator are set at varied pressures up to and including 1100 psig nominal.

<INSERT B3.7-2D>

The LCO is modified by a Note that allows all but one MSSV on each main steam header to be gagged and the setpoints for the two (one on each header) OPERABLE MSSVs to be reset for the duration of hydrotesting in MODE 3. This is necessary to allow the hydrotest pressure to be attained.

<INSERT B3.7-2E>

In MODE 3, the MSSVs satisfy Criterion 4 of 10 CFR 50.36.

BASES

APPLICABILITY (continued)

~~accompanying LCO. Below [18% RTP]~~ In MODES 1, 2, and 3, ~~only~~ the two MSSVs are required to be OPERABLE ~~per steam generator~~ to prevent overpressurization of the main steam system. 1

In MODES 4 and 5, there is no credible transient requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in ~~these~~ MODES. edit

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets ASME Code requirements for the power level. Operation may continue, provided the ALLOWABLE THERMAL POWER and RPS nuclear overpower trip setpoint are reduced as required by Table 3.7.1-1. These values are based on ~~by the application of~~ the following formulas: 1

$$RP = \frac{Y}{Z} \times 100\%$$

and

$$SP = \frac{Y}{Z} \times W$$

where:

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1, "Reactor Protection System (RPS)";

Y = Total OPERABLE MSSV relieving capacity per steam generator based on a summation of individual OPERABLE MSSV relief capacities per steam generator ~~[lb/hour]~~

the available capacity of each MSSV is 801,428 lbm/hour

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Z = Required relieving capacity per steam generator of ~~6,585,600 lb/hour~~; 5,610,000 lbm/hr

RP = Reduced power requirement (not to exceed RTP); and

SP = Nuclear overpower trip setpoint (not to exceed W).

~~These equations are graphically represented in Figure 3.7.1-1, in the accompanying LCO. Operation is restricted to the area below and to the right of line BCDE.~~

~~The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.~~

1

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional ~~8~~ 32 hours is allowed in Required Action A.2 to reduce the setpoints ~~in recognition of the difficulty of resetting of all channels of this trip function within a period of 4 hours~~. The Completion Time of ~~12~~ 36 hours for Required Action A.2 is based on a reasonable Time to correct the MSSU inoperability, the time required to perform the power reduction, on operating experience in resetting all channels of a protective function~~s~~ and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

2

B.1 and B.2

With one or more ~~MSSVs inoperable, a verification by administrative means that at least (two) required MSSVs per steam generator are OPERABLE, with each valve from a different lift setting range, is performed.~~ steam generators with less than two MSSUs OPERABLE, or if the Required Actions and

~~If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time~~ are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating

21

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of ~~each~~ MSSV lift setpoints in accordance with the Inservice Testing Program. The ~~ASME Code, Section XI (Ref. 4) requires that~~ safety and relief valve tests ~~be~~ are performed in accordance with ANSI/ASME OM-1-1987 (Ref. ~~5~~ 6) and include ~~According to Reference 5,~~ the following ~~tests are required~~ for MSSVs: edit

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity ~~device~~ on balanced valves. edit

The ANSI/ASME Standard requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference ~~5~~ 6 provides the activities and ~~frequencies necessary~~ to satisfy the requirements, and ~~Table 3.7.1-1~~ allows an as-found ± ~~1~~ 3% setpoint tolerance. ~~for OPERABILITY; however~~ Although not required by the IST Program, the valves are reset to ± 1% during the Surveillance to allow for drift. edit 1

This SR is modified by ~~a~~ Note 1 that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

INSERT B3.7-5A 1

(continued)

<INSERT B3.7-5A>

The SR is also modified by Note 2 to allow the setpoints for the OPERABLE MSSVs to be reset (one on each header) for the duration of hydrotesting in MODE 3. The remaining valves on each main steam header are typically gagged (or the setpoints may also be reset) to allow the hydrotest pressure to be attained.

BASES (continued)

edit

REFERENCES

1. ~~F~~SAR, Section ~~[5.2]~~. 10.3.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. ~~F~~SAR, ~~Section [15.2]~~. Chapter 14.
~~4. ASME, Boiler and Pressure Vessel Code, Section XI.~~
4. Framatome Document 86-1266156-00, "ANO-1 Overpressure Protection," dated October 31, 1997. (21)
5. 10 CFR 50.36. (6)
6 ~~5~~. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

main steam

The MSIVs isolate steam flow from the secondary side of the steam generators following a ~~high energy~~ line break ~~(HELB)~~. MSIV closure terminates flow from the unaffected (intact) steam generator. edit

One MSIV is located in each main steam line outside of, but close to, ~~containment~~ the reactor building. The MSIVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Turbine Bypass System, and other auxiliary steam supplies from the steam generators.

main steam line isolation (MSLI)

The MSIVs close on a ~~steam and feedwater Rupture Control System~~ signal ~~generated by either low steam generator pressure or steam generator to feedwater differential pressure. The MSIVs fail closed on loss of control or actuation power.~~ ⟨INSERT B3.7-7A⟩ The MSIVs may also be actuated manually. 22

~~A description of the MSIVs is found in the FSAR, Section [10.3] (Ref. 1).~~ edit

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the ~~containment~~ analysis for the ~~large~~ steam line break (SLB) ~~inside containment~~, as discussed in the ~~F~~SAR, Section ~~[6.2]~~ 14.2 (Ref. ~~2~~ 1). ~~It is also influenced by the accident analysis of the SLB events presented in the FSAR, Section [15.4] (Ref. 3).~~ The EFIC System design precludes the blowdown of more than ~~one steam generator, assuming a single active component failure (i.e., the failure of one MSIV to close on demand).~~ as discussed in the SAR, Section 7.1.4 (Ref. 2), edit 22

~~The limiting case for the containment analysis is the SLB inside containment with a loss of offsite power following turbine trip and failure of the MSIV on the affected steam generator to close. At 100% RTP, the steam generator inventory and temperature are at their maximum, maximizing the mass and energy release to the containment.~~

(continued)

<INSERT B3.7-7A>

as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." The EFIC System is designed to prevent the simultaneous blowdown of both steam generators.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Due to reverse flow, failure of the MSIV to close contributes to the total release of the additional mass and energy in the steam headers downstream of the other MSIV. Other failures considered are the failure of a main feedwater isolation valve to close, and failure of an emergency diesel generator (EDG) to start.~~

~~The accident analysis compares several different SLB events against different acceptance criteria.~~ The ~~large~~ SLB outside ~~containment~~ *the reactor building* upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The ~~large~~ SLB ~~inside containment~~ at full power is the limiting case for a post trip return to power. ~~The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.~~ With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. ~~With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed. Significant single failures considered include failure of an MSIV to close, failure of an EDG, and failure of an HPI pump.~~ 22

The MSIVs serve only a safety function *in the event of an SLB* and remain open during power operation. ~~These valves operate under the following situations:~~ *Closing*

~~a. An HELB, an SLB, or main feedwater line breaks (FWLBs), inside containment. In order to maximize the mass and energy release into the containment, the analysis assumes the MSIV in the affected steam generator remains open. For this scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIV in the intact loop.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. ~~An SLB outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.~~

c. ~~A break downstream of the MSIVs will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves will also terminate on closing the MSIVs.~~

d. ~~Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generator. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSIVs' setpoints, a necessary step toward isolating flow through the rupture.~~

e. ~~The MSIVs are also utilized during other events such as an FWLB.~~

22

In MODES 1 and 2, the MSIVs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 3). 6

<INSERT B 3.7-9A>

LCO

This LCO requires that For an ~~the~~ MSIV to be in ~~both~~ each steam line~~s~~ be OPERABLE. ~~The~~ MSIV~~s~~ ~~are~~ considered OPERABLE ~~when~~ the isolation time~~s~~ ~~are~~ must be within limits and the~~y~~ MSIV must close on an isolation actuation signal when required. 22 8

This LCO provides assurance that the MSIVs will perform their design safety function to ~~mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 4).~~ Isolate an SLB. 22

APPLICABILITY

The MSIVs must be OPERABLE in ~~MODE 1 and in~~ MODES 1, 2, and 3 ~~with any MSIVs open,~~ when there is significant mass and energy in the RCS and steam generators ~~; therefore, the MSIVs must be OPERABLE or closed. When the MSIVs are closed, they are already performing the safety function.~~ to provide isolation of potential main steam line breaks 3

(continued)

<INSERT B3.7-9A>

In MODE 3, the MSIVs satisfy Criterion 4 of 10 CFR 50.36.

BASES

APPLICABILITY (continued)

In MODE 4, the steam generator energy is low. Therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators are depressurized and ~~do not contain much energy because their temperature is below the boiling point of water; therefore,~~ the MSIVs are not required for isolation of potential main steam line ~~high energy secondary system pipe~~ breaks ~~in these MODES~~. edit

ACTIONS

A.1

With one or more MSIVs inoperable in MODE 1 or 2, action must be taken to restore the component to OPERABLE status within ~~[8]~~ 24 hours. ~~Some repairs can be made to the MSIV with the unit hot.~~ The ~~[8]~~ 24 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. Although not credited, the ~~The~~ turbine ~~stop~~ throttle valves ~~are~~ may be available to provide ~~the required~~ some isolation for ~~the~~ postulated accidents. 4 edit

INSERT B3.7-10A

~~The [8] hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.~~ 22 edit

B.1

If the Required Action and associated Completion Time of Condition A are not met, ~~MSIV cannot be restored to OPERABLE status within [8] hours,~~ the unit must be placed in MODE ~~2~~ 3 ~~and the inoperable MSIV closed~~ within the next ~~6~~ 12 hours. The Completion Time~~s~~ ~~are~~ is reasonable, based on operating experience, to reach MODE ~~2~~ 3. 5 22 4

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODE~~S~~ ~~2~~ ~~and~~ 3, the inoperable MSIV~~(s)~~ may either be restored to 4

(continued)

<INSERT B3.7-10A>

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

BASES

ACTIONS

C.1 and C.2 (continued)

OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

INSERT B3.7-11A

~~The [8] hour Completion Time is consistent with that allowed in Condition A.~~ 4

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, ~~based on engineering judgment,~~ in view of MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position. edit

D.1 and D.2

If the Required Actions and associated Completion Times of Condition C are not met, ~~MSIV cannot be restored to OPERABLE status or closed in the associated Completion Time,~~ the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed ~~in at least MODE 3 within 6 hours and~~ in MODE 4 within ~~12~~ 24 hours. The allowed Completion Time ~~s are~~ is reasonable, based on operating experience, to reach the required unit conditions from MODE ~~2~~ 3 conditions in an orderly manner and without challenging unit systems. 22 4

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the ~~MSIV~~ isolation closure time of each MSIV is as specified in the Inservice Testing Program ~~≤ [6] seconds on an actual or simulated actuation signal~~. ~~The MSIV closure~~ time is assumed in the accident and ~~containment~~ reactor building analyses. This Surveillance is normally performed ~~upon~~ prior to returning the unit to ~~operation~~ power operation, e.g., during MODE 3, following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. ~~As the MSIVs are not to be tested at power, they are exempt from~~ 8 edit edit edit

(continued)

<**INSERT B3.7-11A**>

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

edit

~~the ASME Code, Section XI (Ref. 5) requirements during operation in MODES 1 and 2.~~

The Frequency for this SR is in accordance with the Inservice Testing Program ~~or [18] months]. The [18] month Frequency to demonstrate the valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

8

edit

normally

This test is conducted in MODE 3, with the unit at operating temperature and pressure~~, as discussed in the Reference 5 exercising requirements.~~ This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

<INSERT B3.7-12A>

REFERENCES

edit

~~1. FSAR, Section [10.3].~~

1 ~~2~~. ~~F~~SAR, Section ~~[6.2]~~ 14.2.

2 ~~3~~. ~~F~~SAR, Section ~~[15.4]~~ 7.1.4.

~~4. 10 CFR 100.11.~~

4 ~~5~~. ASME, Boiler and Pressure Vessel Code, Section XI.

3. 10 CFR 50.36.

6

<INSERT B3.7-12A>

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The Frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.2.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

 9

B 3.7 PLANT SYSTEMS

B 3.7.3 [Main Feedwater ~~Stop~~ Isolation Valves (MF~~S~~IVs) ~~[Main Feedwater Control Valves (MFCVs), and Associated Startup Feedwater Control Valves (SFCVs)]~~ 9

BASES

BACKGROUND

The main feedwater isolation valves (MFIVs) ~~for each steam generator consist of the MFSVs, MFCVs, and the SFCVs. The MFIVs~~ isolate main feedwater (MFW) flow to the secondary side of the steam generators ~~following a high energy line break (HELB)~~. ~~Closure of the MFIVs terminates flow to both steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the feedwater lines downstream of the MFIVs will be mitigated by their closure.~~ Closing the MFIVs ~~and associated bypass valves~~ effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment and reducing the cooldown effects for SLBs. 9 23

edit

The MFIVs close on receipt of a main steam line isolation (MSLI) ~~Steam and Feedwater Rupture Control System (SFRCS)~~ signal ~~generated by either low steam generator pressure or steam generator/feedwater differential pressure~~. The MFIVs can also be closed manually. 23

<INSERT B3.7-13A>

~~The MFIVs and associated bypass valves close on receipt for a safety injection low T_{avg} coincident with reactor trip or steam generator water level high high signal. They may also be actuated manually. In addition to the MFIVs and associated bypass valves, a check valve inside containment is available to isolate the feedwater line penetrating containment and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems.~~ 9

~~A description of the MFIVs is found in the FSAR, Section [10.4.7] (Ref. 1).~~

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the ~~large~~ SLB. ~~It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs may also be relied on to terminate a steam break for core response~~ 23

(continued)

<INSERT B3.7-13A>

as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation."

9

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~analysis and excess feedwater event upon the receipt of a steam generator water level high signal.~~ 23

Failure of an MFIV to close following an SLB~~, FWLB, or excess feedwater event~~ can result in additional mass ~~and energy~~ being delivered to the steam generators, contributing to cooldown. ~~This failure also results in additional mass and energy releases following an SLB or FWLB event.~~

The MFIVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 1). 6

LCO

In MODES 1 and 2,

<INSERT B3.7-14A>

edit

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators following ~~a FWLB or~~ a main steam line break. ~~These valves will also isolate the nonsafety related portions from the safety related portions of the system.~~ 9

Two MFIVs ~~[Two] [MFSVs], [MFCVs], [or associated SFCVs]~~ are required to be OPERABLE. For an ~~The~~ MFIV~~s~~ ~~are~~ to be considered OPERABLE ~~when~~ the isolation times must be ~~are~~ within limits and the~~y~~ MFIV must close on an isolation actuation signal. 23

Failure to meet the LCO requirements can result in a more severe cooldown transient and in additional mass and energy being released to ~~containment~~ reactor building following an SLB ~~or FWLB~~ inside ~~containment~~. ~~If the SFRCS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.~~ When required. 23

APPLICABILITY

The MFIVs ~~[MFSVs], [MFCVs], [or associated SFCVs]~~ must be OPERABLE in MODES 1, 2, and 3 to ~~whenever there is significant mass and energy in the RCS and steam generators.~~ ~~This~~ ensures that, in the event of an ~~SLB~~ ~~HELB~~, ~~a single failure cannot result in the blowdown of more than one steam generator.~~ the amount of feedwater provided to the affected steam generator is limited. Their closure terminates normal feedwater flow to limit the overcooling transient and 9

~~In MODES 1, 2, and 3, the [MFSVs], [MFCVs], [or associated SFCVs] are required to be OPERABLE in order to limit the~~ amount of ~~available fluid~~ energy that could be added to ~~containment~~ reactor building in the case of a secondary system pipe break inside ~~containment. When the valves are closed, they are already performing their safety function.~~ reactor building 9 3

(continued)

<INSERT B3.7-14A>

In MODE 3, the MFIVs satisfy Criterion 4 of 10 CFR 50.36.

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the ~~[MFSVs], [MFCVs], [or associated SFCVs]~~ MFIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES. 9

ACTIONS

Condition C ~~The ACTIONS table~~ is modified by a Note indicating that separate Condition entry is allowed for each ~~valve~~ MFIV. 9

4

INSERT B3.7-15A

C~~A~~.1 and C~~A~~.2 9

With one ~~[MFSV] in one~~ or more ~~flow paths~~ MFIVs inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within ~~[8 or 72]~~ 48 hours. ~~When these valves are closed or isolated, they are performing their required safety function.~~ attempt minor repairs, and if unsuccessful, to 4

~~For units with only one MFIV per feedwater line:~~ The ~~[8] hour~~ Completion Time is reasonable to close the MFIV ~~or~~ a isolate ~~its associated bypass valve which includes performing a controlled unit shutdown to MODE 2. The Completion Time is reasonable, based on operating experience, to reach MODE 2 from full power conditions with the MFIVs closed, in an orderly manner and without challenging unit systems.~~ e

INSERT B3.7-15B 6

~~The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The [72] hour Completion Time is reasonable, based on operating experience.~~ 4

Inoperable ~~[MFSVs]~~ MFIVs that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. ~~This is necessary to ensure that the assumptions in the safety analysis remain valid.~~ The 7 day Completion Time is reasonable~~, based on engineering judgment,~~ in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated. edit

(continued)

<INSERT B3.7-15A>

A.1

With one or more MFIVs inoperable in MODE 1 or 2, action must be taken to restore the MFIV(s) to OPERABLE status within 24 hours. Some repairs can be made to the MFIV with the unit hot. The 24 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MFIVs occurring during this time interval, and the isolation capability provided by the main feedwater block and control valves. Under normal conditions the main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MFIV problems, thereby avoiding unnecessary shutdown.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in MODE 3 within the next 12 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 3. Further, the unit must be placed in MODE 3 prior to MFIV closure to preclude an undesired transient.

<INSERT B3.7-15B>

Under normal conditions the main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MFIV problems, thereby avoiding unnecessary shutdown. Isolation of an inoperable MFIV may be accomplished by closing the main feedwater block, low load control, and startup control valves (SAR, Table 10-1 (Ref. 2)).

MFIVs

9

BASES

ACTIONS (continued)

B.1 and B.2

With one [MFCV] in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within [8 or 72] hours. When these valves are closed or isolated, they are performing their required safety function.

9

[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable, based on operating experience, to close the MFIV or its associated bypass valve.]

The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable [MFCVs] that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

With one [SFCV] in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within [8 or 72] hours. When these valves are closed or isolated, they are performing their required safety function.

[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable, based on operating experience, to close the MFIV or its associated bypass valve.]

The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the

(continued)

BASES

ACTIONS

~~C.1 and C.2 (continued)~~

~~low probability of an event occurring during this time period that would require isolation of the MFW flow paths.~~

~~Inoperable SFCVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.~~ 9

~~D.1~~

~~With two inoperable valves in the same flow path there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure to two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path and as such is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to close the MFIV or otherwise isolate the affected flow path.~~

D [E].1 ~~and E.2~~ 23

If the Required Actions and associated Completion Times of Condition C are not met, ~~[MFSVs], [MFCVs], and associated SFCVs] cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time,~~ the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed ~~in at least MODE 3 within 6 hours~~ 4 ~~and~~ in MODE 4 within [12] hours. The allowed Completion Time~~s~~ ~~are~~ is reasonable, based on operating experience, to reach the 24 required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

 9

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV ~~[MFSV], [MFCV], and [associated SFCV]~~ is ~~≤ 7 seconds on an actual or simulated actuation signal.~~ as specified in the Inservice Testing Program. 9

The MFIV ~~[MFSV], [MFCV], and [associated SFCV]~~ ~~closure~~ isolation time is assumed in the accident and ~~containment~~ reactor building analyses. This Surveillance is normally performed ~~upon~~ prior to returning the unit to ~~operation~~ power operation following a refueling outage. The ~~[MFSV]~~, MFIVs ~~[MFCV], and [associated SFCV] should~~ are not ~~be~~ tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. ~~This is consistent with the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.~~ e.g. during MODE 3, 9 8

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

The Frequency for this SR is in accordance with the ~~[~~Inservice Testing Program~~, or [18] months]. The Frequency of [18] months for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency.~~

<INSERT B3.7-18A>

REFERENCES

1. 10 CFR 50.36.
2. ~~F~~SAR, ~~Section [10.4.7].~~ Table 10-1. edit
~~2. ASME, Boiler and Pressure Vessel Code, Section XI.~~ edit 6

<INSERT B3.7-18A>

This SR verifies that each MFIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.3.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when the steam generator pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

11

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Vent Valves (AVVs)

BASES

BACKGROUND

The AVVs provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the FSAR, Section [10.3] (Ref. 1). This is done in conjunction with the Emergency Feedwater System, providing cooling water from the condensate storage tank (CST). The AVVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Turbine Bypass System.

[The AVVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation.]

The AVVs are equipped with pneumatic controllers to permit control of the cooldown rate.

[The AVVs are provided with a pressurized gas supply of bottled nitrogen that, on loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the AVVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the AVVs for the time required for Reactor Coolant System (RCS) cooldown to DHR entry conditions.]

A description of the AVVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the AVVs is established by the capability to cool the unit to MODE 3. The design rate of [75]°F per hour is applicable for both steam generators, each with one AVV. This rate is adequate to cool the unit to DHR entry conditions with only one AVV and one steam generator utilizing the cooling water supply available in the CST.

In the accident analysis presented in Reference 1, the AVVs are assumed to be used by the operator to cool down the unit

~~(continued)~~

11

BASES

APPLICABLE SAFETY ANALYSES (continued)

to MODE 3 for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the AVVs and the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator's pressure and temperature below the design value. This is about 30 minutes following initiation of an event; however, this may be less for a steam generator tube rupture (SGTR) event. Some initiating events falling into this category are a main steam line break upstream of the main steam isolation valves, a feedwater line break, and an SGTR event (although the AVVs on the affected steam generator may still be available following an SGTR event).

For the recovery from an SGTR event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to DHR conditions for this event, and also for other accidents. Thus, the SGTR is the limiting event for the AVVs. The number of AVVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the consideration of any single failure assumptions regarding the failure of one AVV to open on demand.

[The design must accommodate the single failure of one AVV to open on demand, thus each steam generator must have at least one AVV. The AVVs are equipped with manual block valves in the event an AVV spuriously fails open, or fails to close during use.]

The AVVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

[Two] AVVs [lines per steam generator] are required to be OPERABLE. Failure to meet the LCO can result in the inability to cool the unit to DHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An AVV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the AVVs are required to be OPERABLE.

In MODES 5 and 6, an SGTR is not a credible event.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

With one AVV [line] inoperable, action must be taken to restore the inoperable AVV to OPERABLE status. The 7 day Completion Time allows for redundant capability afforded by the remaining OPERABLE AVV and a nonsafety grade backup in the Steam Bypass System and MSSVs.

[B.1

With more than one AVV [line] inoperable, action must be taken to restore [all but one] AVV [lines] to OPERABLE status. As the block valve can be closed to isolate an AVV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable AVV [lines], based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the AVV [lines].]

C.1 and C.2

If the AVV [lines] cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours, without reliance upon the steam generator for heat removal. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

(11)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the AVVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the AVVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an AVV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

[SR 3.7.4.2

The function of the block valve is to isolate a failed open AVV. Cycling the block valve closed and open demonstrates its ability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.]

REFERENCES

1. FSAR, Section [10.3].

Generic term. change.
CST → QCST

edit

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The safety related EFW pumps take suction ~~through separate and independent suction lines~~ from the Q condensate storage tank (CST) (LCO 3.7.6, "Q Condensate Storage Tank (CST)"), and pump to the steam generator secondary side through the EFW nozzles. The ~~steam generators function as a heat sink for~~ core decay ~~heat. The~~ heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)"), or atmospheric ~~vent~~ dump valves (~~AVVs~~ ADVs) ~~(LCO 3.7.4, "Atmospheric Vent Valves (AVVs)")~~. If the main condenser is available, steam may be released via the Turbine Bypass valves. ~~System and recirculated to the CST.~~ 25 edit 11

edit

~~The following system description is provided as an example. Actual system description should be provided by the specific unit.~~ The EFW System ~~consists of~~ includes ~~two~~ one turbine driven EFW pumps, ~~each of which provides a nominal 100% capacity,~~ and one ~~non~~safety grade motor driven EFW pump. The ~~steam~~ turbine driven EFW pump~~s~~ receives steam from either of the two main steam headers, upstream of the main steam isolation valves (MSIVs). The EFW System supplies a common header capable of feeding either or both steam generators. ~~The~~ Either pump ~~100% capacity~~ is sufficient to remove decay heat and cool the unit to decay heat removal (DHR) entry conditions. The EFW System ~~normally~~ initially receives a supply of water from the CST. ~~A~~ The assured safety grade source of water is ~~also~~ supplied by the Service Water System (SWS). ~~Automatic~~ Valves on the supply piping are manually opened ~~open on low pressure in the supply piping~~ to transfer the water supply from the CST to the SWS. ~~A third source of~~ water can be supplied from other sources by manually aligning ~~the fire protection header to the EFW pump suction.~~ nonsafety grade condensate storage tanks to the EFW pump suction. Thus, ~~the requirements for~~ diversity in motive power sources for the EFW System ~~are met~~ is provided. 25

The EFW System is capable of supplying feedwater to the steam generators during normal unit startup and shutdown evolutions, and during hot standby conditions, if required. However, EFW does not provide a normal source of feedwater during these conditions. The normal supplement to the main feedwater system under these conditions is provided by the auxiliary feedwater system. 25

(continued)

BASES

BACKGROUND (continued)

~~The EFW System is designed to supply sufficient water to cool the unit to DHR entry conditions with steam being released through the ADVs or condenser.~~

The EFW actuates automatically on low steam generator level, low steam generator pressure, or loss of four reactor coolant pumps (e.g., on loss of main feedwater pumps,).

The EFW System is discussed in the FSAR, Sections ~~[9.2.7]~~ 7.1.4 and ~~[9.2.8]~~ 10.4.8 (Refs. 1 and 2, respectively).

<INSERT B3.7-24A>

APPLICABLE SAFETY ANALYSES

The EFW System ~~mitigates the consequences of any event with a loss of normal feedwater.~~ is sized to prevent exceeding 110% RCS design pressure for a specified loss of feedwater scenario (Ref. 3)

The design basis of the EFW System is to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure ~~plus 3%~~.

~~In addition, the EFW System must supply enough makeup water to replace steam generator secondary inventory being lost as steam as the unit cools to MODE 4 conditions. Sufficient EFW flow must also be available to account for flow losses such as pump recirculation and line breaks.~~

~~The limiting Design Basis Accidents (DBAs) and transients for the EFW System are as follows:~~

~~a. Feedwater line break (FWLB); and~~

~~b. Loss of main feedwater.~~

~~In addition, the minimum available EFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.~~

The EFW System design is such that it can perform its function with only one EFW train available ~~following a loss of the turbine driven main feedwater pumps or an FWLB, combined with a loss of normal or reserve electric power~~.

IN MODES 1 and 2, the EFW System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 4).

In MODE 3 and MODE 4 when steam generator(s) are relied upon for heat removal, the EFW System satisfies Criterion 4 of 10CFR50.36.

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25
25
25
6

(continued)

<INSERT B3.7-24A>

as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation."

BASES (continued)

LCO

This LCO provides assurance that the EFW System will perform its design ~~safety~~ function to mitigate the consequences of ~~accidents~~ events that could result in overpressurization of the reactor coolant pressure boundary. ~~[Three]~~ Two independent ~~EFW pumps, in two diverse~~ trains are required to be OPERABLE to ensure the availability of residual heat removal capability. ~~for all events accompanied by a loss of offsite power and a single failure.~~ ~~[This is accomplished by powering two pumps by steam driven turbines supplied with steam from a source not isolated by the closure of the MSIVs, and one pump from a power source that, in the event of loss of offsite power is supplied by the emergency diesel generator.]~~

Two 12 25 edit

For both EFW trains ~~The EFW System~~ ~~is~~ to be considered ~~to be~~ OPERABLE, ~~when~~ the components and flow paths required to ~~provide~~ be capable of providing EFW flow to both the steam generators ~~are OPERABLE~~. This requires that the ~~[two]~~ turbine driven EFW pump ~~(s)~~ be OPERABLE with ~~redundant~~ two steam supplies (one from each of the main steam lines upstream of the MSIVs) and capable of supplying EFW flow to ~~either of~~ the ~~two~~ steam generators. The ~~[non]~~safety grade motor driven EFW pump ~~(s)~~ is ~~and other associated flow paths to the EFW System [are]~~ also required to be OPERABLE and capable of supplying EFW flow to the steam generators. The piping, valves, instrumentation, and controls in the required flow paths ~~shall~~ must also be OPERABLE. ~~The primary and secondary sources of water to the EFW~~ System are required to be OPERABLE. The associated flow paths from the EFW System primary and secondary sources of water to ~~all~~ both EFW pumps also are required to be OPERABLE.

edit 25 edit edit

The LCO is modified by a Note indicating that only one EFW train, which includes ~~a~~ the motor driven EFW pump, is required in MODE 4. This is because of reduced heat removal requirement, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

edit

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE ~~and~~ in order to function in the event that the main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

edit

(continued)

BASES

APPLICABILITY (continued)

are relied upon for decay heat removal since EFW is the safety related source of feedwater to the steam generators.

must be OPERABLE when

In MODE 4, ~~with RCS temperature above [212]°F,~~ the EFW System ~~may be used for heat removal via~~ the steam generators. In MODE 4, the steam generators are normally used for heat removal until the DHR System is in operation. 25

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

ACTIONS

A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, action must be taken to restore the steam supply to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The availability of the redundant OPERABLE steam supply to the turbine driven EFW pump~~(s)~~; edit

b. The availability of the redundant OPERABLE motor driven EFW pump; and

c. The low probability of an event occurring that would require ~~the inoperable steam supply to~~ use of the turbine driven EFW pump~~(s)~~. edit

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of ~~Conditions~~ required EFW components to be inoperable during any continuous failure to meet this LCO. edit

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 ~~2 hours~~ days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met. edit edit

B.1

When one of the required EFW trains (pump or flow path) is inoperable, action must be taken to restore the train to

(continued)

BASES

ACTIONS

B.1 (continued)

OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to ~~one of~~ the turbine driven EFW pump~~s~~. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of ~~a DBA~~ an event requiring EFW occurring during this time period. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of ~~Conditions~~ required EFW components to be inoperable during any continuous failure to meet this LCO. edit

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met. edit

C.1 and C.2

With the ~~When either~~ Required Action ~~B.1 or Required Action B.1 cannot be completed within the required Completion Time, [or when two EFW trains are inoperable in MODE 1, 2, or 3,]~~ and associated Completion Time of Condition A or B not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within ~~[~~18~~]~~ hours. 25

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

~~In MODE 4, with two EFW trains inoperable, operation is allowed to continue because only one motor driven EFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate DHR.~~ 12

(continued)

BASES

ACTIONS (continued)

D.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW train is restored to OPERABLE status.

With ~~all~~ both EFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit~~s~~ into a less safe condition. edit

E.1

In MODE 4, either the steam generator the required loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops MODE 4." With ~~one~~ EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. edit

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. Correct alignment for automatic valves may be other than the post-accident position provided the valve is otherwise OPERABLE. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. 25

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

The 31 day Frequency is based on ~~engineering judgment, is consistent with~~ the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the established acceptance criteria during the cycle. Flow and differential head are ~~normal tests~~ indicators of pump performance required by Section XI of the ASME Code (Ref. 5). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test ~~is~~ may be performed on ~~recirculation flow.~~ a test flow path. edit edit edit

This test ~~confirms one point on the pump design curve and~~ is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 5), ~~at 3 month intervals~~, satisfies this requirement. ~~The [31] day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 3.~~ edit 13

This SR is modified by a Note indicating that the SR ~~should~~ may be deferred until suitable test conditions are established. This deferral is required because there ~~is~~ may be insufficient steam pressure to perform the test. edit edit

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in an Emergency the event of any accident or transient that generates ~~a Steam and~~ Feedwater ~~Rupture~~ Initiation and Control System (EFIC) ~~(SFRCS)~~ signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Each automatic valve is also verified to be capable of manual operation by over-riding the actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a 25

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.3 (continued)

edit

unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is also acceptable based on operating experience and design reliability of the equipment. ~~This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is already aligned and operating.~~ This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] states which that the SR is not required in MODE 4 when the steam generator is being relied upon for heat removal. [In MODE 4, the required pump is already operating and the autostart function is not required.] [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.]

¶

edit

14

SR 3.7.5.4

This SR verifies that ~~the turbine driven~~ each EFW pump starts in the event of any accident or transient that generates an EFIC ~~SFRCS~~ signal by demonstrating that each turbine driven EFW pump starts automatically on an actual or simulated actuation signal. ~~These pumps are not required in MODE 4.~~ The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] which states that the SR is not required in MODE 4. [In MODE 4, the required pump is already operating and the autostart function is not required.] [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.]

25

14

edit

edit

14

~~Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.~~

25

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Any combination of MODE 5 or 6 or defueled. 47

SR 3.7.5.5

This SR ensures that the EFW System is properly aligned by verifying the position of manual valves in the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in ~~MODE 5 or 6~~. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable~~, based on engineering judgment,~~ in view of other administrative controls, such as SR 3.7.5.1, to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of manual valves has occurred. This SR ensures that the flow path from the CST to the steam generator is properly aligned. ~~(This SR is not required by those units that use EFW for normal startup and shutdown.)~~ edit 15 Q 46

<INSERT B 3.7-31A>

~~SR 3.7.5.6 and SR 3.7.5.7~~

~~For this facility, the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the EFW pump suction pressure interlocks are as follows:~~ 16

REFERENCES

1. ~~F~~SAR, Section ~~9.2.7~~, 7.1.4. edit
2. ~~F~~SAR, Section ~~9.2.8~~, 10.4.8. edit
3. NRC Letter dated January 12, 1981, (1CNA018103). 25
4. 10 CFR 50.36. 6
5. ~~3~~. ASME, Boiler and Pressure Vessel Code, Section XI. edit

<INSERT B3.7-31A>

SR 3.7.5.6

This SR ensures that the EFW flowpath to each steam generator is open and that water reaches the steam generators from the EFW System. This test is performed during shutdown to minimize thermal cycles to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater. The motor-driven EFW pump is specified because of its availability at the low steam generator pressure conditions that exist in the shutdown condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Generic term. change, CST → QCST. — every use —

edit

B 3.7 PLANT SYSTEMS

B 3.7.6 Q Condensate Storage Tank ~~(CST)~~ (QCST)

condensate storage tank (QCST)

BASES

BACKGROUND

The ~~CST~~ provides a ~~safety grade~~ source of demineralized water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides ~~a passive flow~~ the preferred source of water ~~by gravity~~ to the Emergency Feedwater (EFW) System (LCO 3.7.5, "Emergency Feedwater (EFW) System"). ~~The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) or the atmospheric vent valves.~~ 26

edit

~~When the main steam isolation valves are open, the preferred means of heat removal is to discharge to the condenser by the nonsafety grade path of the turbine bypass valves. The condensed steam is returned to the CST by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.~~ 26

Because the CST is ~~a principal component in removing residual heat from the RCS~~ the normally aligned source to EFW, it is designed to withstand earthquakes and other natural phenomena, and a portion is protected from ~~as well as~~ missiles that might be generated by natural phenomena. The CST is designed as ~~to~~ Seismic Category I to ensure availability of the initial EFW ~~feedwater~~ supply. Feedwater is also available from ~~an~~ alternate sources ~~(s)~~. 26

A description of the CST is found in the ~~F~~SAR, Section ~~[9.2.6]~~ 10.4.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides the initial source of cooling water to remove decay heat and cool down the unit following ~~all~~ any events with a loss of normal feedwater. ~~in the accident analysis, as discussed in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs, followed by a cooldown to decay heat removal (DHR) entry conditions at the design cooldown rate.~~ 26

~~The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

power. Single failures that also affect this event include the following:

a. Failure of the diesel generator powering the motor driven EFW pump to the unaffected steam generator (requiring additional steam to drive the remaining EFW pump turbine); and

<INSERT B3.7-33A>

b. Failure of the steam driven EFW pump (requiring a longer time for cooldown using only one motor driven EFW pump).

These are not usually the limiting failures in terms of consequences for these events. (26)

The CST satisfies Criterion ~~2~~ 4 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2). (6)

LCO

To satisfy accident analysis assumptions, the [two] CSTs must contain sufficient cooling water to remove decay heat for 13 hours following a reactor trip from 102% RTP and then to cool down the RCS to DHR System entry conditions, assuming a coincident loss of offsite power and most adverse single failure. While so doing, the CSTs must retain sufficient water to ensure adequate net positive suction head for the EFW pump(s) during the cooldown, to account for any losses from the steam driven EFW pump turbine, as well as losses incurred before isolating EFW to a broken line. (26)

<INSERT B3.7-33B>

The level required is equivalent to a usable volume of [250,000] gallons, which is based on holding the unit in MODE 3 for 13 hours, followed by a cooldown to DHR System entry conditions.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, 3, and in MODE 4~~,~~ (a) when a steam generator is being relied upon for heat removal, the CST is required to be OPERABLE. edit

In MODE 4 when a steam generator is not being relied upon for heat removal and in MODES 5 and 6, the CST is not required because the EFW System is not required. (26)

(continued)

<INSERT B3.7-33A>

A portion of the QCST (T-41B) is protected from tornado generated missiles. The protected volume is sufficient to provide a thirty minute supply of water which is adequate to allow manual operator action, if required, to transfer suction of the EFW pumps to service water.

<INSERT B3.7-33B>

The OPERABILITY of the QCST with the minimum required water volume ensures that sufficient water is available to support EFW operation on both units for at least 30 minutes. This provides adequate time fir the operators to manually switch the EFW suction alignment to the Service Water System (SWS), if required. The SWS provides the assured long-term source of cooling water. The required volume considers that the EFW suctions of both units may be aligned to the QCST simultaneously.

The required minimum usable volume includes an allowance for losses due to Unit 2 recirculation line flow. The required volume of 32,300 gallons is equivalent to a tank level of 3 feet 10 inches. This parameter value does not include allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

The tank has sufficient capacity to support more than four hours of cooling in MODE 3 or MODE 4 conditions for both units. This capability is not considered to be a safety related design function and is not controlled by the Technical Specifications.

BASES (continued)

ACTIONS

A.1 and A.2

As an alternative to unit shutdown, the OPERABILITY of the backup water supply should be verified within 4 hours and once every 24 hours thereafter. The OPERABILITY of the backup feedwater supply must include verification, by administrative means, of the OPERABILITY of flow paths from the backup supply to the EFW pumps and availability of the required volume of water in the backup supply. The CST must be restored to OPERABLE status within 7 days because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period, requiring the use of the water from the CST(s).

12

Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available.

48

48

edit

B.1 and B.2

Required Action and

If the ~~CST cannot be restored to OPERABLE status in the~~ associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generators for heat removal, within ~~[18 hours]~~. ~~This allows an additional 6 hours for the DHR System to be placed in service after entering MODE 4.~~

s are not met

9

26

edit

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST(s) contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including

edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1 (continued)

alarms, to alert the operator to abnormal deviations in CST levels.

REFERENCES

1. ~~F~~SAR, Section ~~[9.2.6]~~. 10.4.8.
2. ~~FSAR, Chapter [6].~~ 10 CFR 50.36.
3. ~~FSAR, Chapter [15].~~

edit
6
edit

17

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment.

A typical CCW System is arranged as two independent full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. A surge tank in the system provides sufficient net positive suction head for each pump and isolation of nonessential components on a low tank level signal. The pump in each train is automatically started on receipt of a safety feature actuation signal, and all nonessential components are isolated.

Additional information on the design and operation of the CCW System, along with a list of the components served, is presented in the FSAR, Section [9.2.2] (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the [decay heat removal (DHR) heat exchanger]. This may utilize the DHR System during a normal or post accident cooldown and shutdown, or during the recirculation phase following a loss of coolant accident.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is to provide cooling water to the Emergency Core Cooling System and emergency diesel generators (EDGs) during DBA conditions. The CCW System also supplies cooling water to EDGs during a loss of offsite power.

(continued)

17

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CCW System is designed to perform its function with a single failure of any active component assuming a loss of offsite power.

The CCW System also functions to cool the unit from [DHR] entry conditions (T_{cold} < [350]°F) to MODE 5 (T_{cold} < [200]°F) during normal and post accident operations. The time required to cool from [350]°F to [200]°F is a function of the number of CCW and [DHR] trains operating. One CCW train is sufficient to remove decay heat during subsequent operations with T_{cold} < [200]°F.

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one train does not depend on the other. In the event of a DBA, one train of CCW is required to provide the minimum heat removal capability assumed in the safety analysis for systems to which it supplies cooling water. To ensure this is met, two CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with loss of offsite power.

A CCW train is considered OPERABLE when:

a. It has an OPERABLE pump and associated surge tank; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render these components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post

(continued)

17

BASES

APPLICABILITY (continued)

accident safety functions, primarily Reactor Coolant System heat removal, by cooling the DHR heat exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources" Operating," and LCO 3.4.6, "RCS Loops" MODE 4," should be entered if an inoperable CCW train results in an inoperable EDG or DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW train cannot be restored to OPERABLE status in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those

(continued)

17

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

components inoperable, but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated

(continued)

(17)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.3 (continued)

as part of routine testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.2].

B 3.7 PLANT SYSTEMS

B 3.7.~~8~~ 7 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a transient or Design Basis Accident (DBA) ~~or transient~~. During normal operation and normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. ~~The safety related portion is covered by this LCO.~~

The ~~An~~ SWS consists of two independent but interconnected, ~~separate,~~ 100% capacity safety related cooling water loops ~~trains~~. Each loop ~~train~~ consists of a ~~100% capacity~~ pump, ~~one component cooling water (CCW) heat exchanger,~~ piping, valving, sluice gates, and instrumentation. The pumps, Three 100% capacity pumps are provided to supply the two trains. and valves are remote manually aligned, ~~except in the unlikely event of a loss of coolant accident (LOCA). The pumps are automatically started upon receipt of a safety feature actuation signal, and all~~ essential valves are aligned to their post accident positions upon receipt of an engineered safeguards actuation signal. The SWS ~~also~~ ~~provides cooling directly to the Control Room Emergency Ventilation System water cooled condensing unit, the Emergency Core Cooling System (ECCS) pump room coolers, containment air cooler, and turbine driven cooling water systems.~~ required plant equipment. The system ~~provides cooling and~~ is also ~~a~~ the assured safety related source of water to the ~~ECCS pump and the~~ emergency feedwater pumps, and can also provide a source of makeup water to the emergency cooling ~~tower~~ pond, and to the spent fuel pool. 27

<INSERT B3.7-41A>

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the ~~U~~SAR, Section ~~9.2.1~~ 9.3 (Ref. 1). The principal safety related function of the SWS is the ~~removal~~ transfer of ~~decay~~ heat from the reactor ~~via the CCW System~~ and safety related components to the heat sink.

APPLICABLE SAFETY ANALYSES

The primary safety function ~~design basis~~ of the SWS is for one SWS train, in conjunction with the Low Pressure Injection System and the ~~CCW System and a 100% capacity~~ ~~containment~~ Reactor Building cooling system, (~~containment~~ reactor building spray, ~~containment~~ Reactor Building air coolers, or a combination) to remove core decay heat following a design basis LOCA, as discussed in the ~~U~~SAR, Sections ~~6.2.1 (Ref. 2).~~ 6.2 and 6.3 (Refs. 2 and 3, respectively). ~~This provides for a gradual reduction in the temperature of this fluid, as it is~~

(continued)

<INSERT B3.7-41A>

The requirements of the service water system for cooling water are more severe during normal operation (at full power) than under accident conditions. Normal operation requires at least two of the three service water pumps, and the pumps in operation are periodically rotated. Normal operation also includes the addition of a biocide during the reactor building emergency cooler surveillance, when the water temperature is between 60°F and 80°F, to prevent biological fouling of the coolers. This water temperature range provides conditions under which Asian clams can spawn and produce larvae which could pass through service water system strainers.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~supplied to the Reactor Coolant System (RCS) by the safety injection pumps.~~

edit

The SWS is designed to perform its function with a single failure of any active component, assuming loss of offsite power.

The SWS~~, in conjunction with the CCW System,~~ also cools the unit from Decay Heat Removal (DHR) System, as discussed in the ~~F~~SAR, Section [6.3], (Ref ~~4~~ 3) entry conditions to MODE 5 during normal and post accident operation. The time required for this evolution is a function of the number of ~~CCW and~~ DHR System ~~trains~~ loops that are operating. ~~One SWS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum SWS temperature of [85]°F occurring simultaneously with maximum heat loads on the system.~~ 9.5 27

The SWS is also required ~~when needed~~ to ~~support CCW in the removal of~~ transfer heat from the ~~emergency~~ diesel generators (EDGs)~~, of reactor auxiliaries~~. edit

In MODES 1 and 2 ~~The SWS satisfies~~ Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 5). 6

In MODES 3 and 4, the SWS satisfies Criterion 4 of 10 CFR 50.36.

edit

LCO

Two SWS ~~trains~~ loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

For ~~An~~ an SWS ~~train~~ loop ~~is~~ to be considered OPERABLE, ~~when:~~ it must have: edit

a. ~~It has an~~ One OPERABLE pump; and 27

b. The associated piping, valves, ~~heat exchanger,~~ sluice gates, and instrumentation and controls required to perform the safety related function ~~are~~ OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS. ~~and~~ Therefore, the SWS is required to be OPERABLE in these MODES. edit

(continued)

BASES

APPLICABILITY (continued)

In MODES 5 and 6, the ~~OPERABILITY~~ requirements of the SWS are determined by the systems it supports. edit

ACTIONS

A.1 edit

If one SWS ~~train~~ loop is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS ~~train~~ is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS ~~train~~ loop could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources — Operating," should be entered if an inoperable SWS ~~train~~ loop results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," should be entered if an inoperable SWS ~~train~~ loop results in an inoperable DHR ~~train~~. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE ~~train~~, and the low probability of a DBA occurring during this period. edit edit edit

B.1 and B.2

If the Required Action and ~~SWS train cannot be restored to OPERABLE status within the~~ associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. edit

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (7)

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~8~~ 7.1 (continued)

otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is ~~based on engineering judgment, is~~ consistent with the existence of procedural controls governing valve operation, and ensures correct valve positions. edit

This SR is modified by a Note indicating that the isolation of ~~the SWS~~ components or systems may render those components inoperable, ~~but~~ does not affect the OPERABILITY of the SWS. supported by the SWS However, such isolation edit

SR 3.7.~~8~~ 7.2

The SR verifies proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The ~~[~~18~~]~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[~~18~~]~~ month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. edit

~~SR 3.7.8.3~~

~~The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the~~ 18

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.7.8.3 (continued)~~

~~Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at an [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

18

REFERENCES

1. SAR, Section ~~9.2.11~~ 9.3.
2. SAR, Section 9.6.2.
3. SAR, Section 9.6.3.
4. SAR, Section 9.5.
5. 10 CFR 50.36.

edit

6

B 3.7 PLANT SYSTEMS

B 3.7.~~9~~ 8 ~~Ultimate Heat Sink (UHS)~~ Emergency Cooling Pond (ECP)

BASES

BACKGROUND

The ~~UHS~~ ECP provides a shared heat sink for ~~process and~~ removing operating heat from safety related components ~~during a transient or accident as well as during normal operation~~ if the heat sink provided by the Dardanelle Reservoir is unavailable. This is done utilizing the Service Water System (SWS).

The ~~UHS has been defined as that~~ ECP is a portion of the complex of water sources~~,~~ which fulfill the ultimate heat sink requirements for ANO. This complex includes the ~~including~~ necessary retaining structures ~~(e.g., a pond with its dam, or a river with its dam)~~, and the ~~canals or~~ piping ~~conduits~~ connecting the sources with, but not including, the ~~cooling water system~~ SWS intake structure~~s~~, as discussed in the ~~U~~SAR, Section ~~9.2.5~~ 9.3 (Ref. 1). ~~If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink.~~ The ~~two~~ principal function~~s~~ of the ~~UHS are~~ ECP is the dissipation of residual heat after a reactor shutdown~~, and dissipation of residual heat after an accident~~. 28

~~A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.~~

The basic performance for both units requirements are that a 30 day supply of water be available~~,~~ and that the design basis temperatures of safety related equipment not be exceeded. ~~Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on another source(s) and a makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations and multiple makeup water sources may be required).~~

Additional information on the design and operation of the system~~, along with a list of components served,~~ can be found in Reference 1.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ~~UHS~~ ECP is the sink for heat removal from the reactor core following ~~all accidents~~ and ~~anticipated operational occurrences~~ abnormality in which the unit is cooled down and placed on decay heat removal following a loss of the Dardanelle Reservoir inventory which would be considered a single failure. ~~Its maximum post accident heat load occurs approximately 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems are required to remove the core decay heat.~~ 51 28

The operating limits are based on conservative heat transfer analyses for the worst case ~~LOCA~~. Reference 1 provides the details of the assumptions used in the analysis. (INSERT B3.7-47A) ~~These assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case failure (e.g., single failure of a manmade structure).~~ (INSERT B3.7-47B) The ~~UHS~~ is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water ~~in the UHS~~ ECP. 28

The ~~UHS~~ ECP satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 6

LCO

The ~~UHS~~ ECP is a backup system that is required to be OPERABLE to support the SWS. To be ~~and is~~ considered OPERABLE, the ECP must ~~if it~~ contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 30 days following the design basis ~~LOCA~~ event without ~~the loss of net positive suction head (NPSH), and without~~ exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the ECP initial ~~UHS~~ temperature should not exceed ~~90~~ 100°F, and the ~~level~~ volume of water should not fall below ~~562 ft (mean sea level)~~ 70 acre-feet during normal unit operation. edit 20

APPLICABILITY

In MODES 1, 2, 3, and 4, the ~~UHS~~ ECP is a ~~normally operating~~ backup system that is required to support the OPERABILITY of the equipment serviced by the ~~UHS~~ SWS and is required to be OPERABLE in these MODES.

In MODES 5 and 6, the ~~OPERABILITY requirements of the UHS are determined by the systems it supports.~~ ECP is not required to be OPERABLE.

(continued)

<INSERT B3.7-47A>

initial conditions that could be present considering a Unit 2 Design Basis Accident concurrent with a normal shutdown of Unit 1 and a loss of the Dardanelle Reservoir water inventory.

<INSERT B3.7-47B>

The minimum ECP requirements take into account: water loss from evaporation due to heat load and climatological conditions, fire pump usage, ECP bottom irregularities, suction pipe level at the ECP, and operator action in transferring the service water system from the Dardanelle Reservoir. Operator action is credited in the inventory analysis during the transfer of the service water system to the ECP. Specifically, pump returns are transferred to the ECP shortly after the Dardanelle Reservoir loss of inventory event begins and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the ECP, lake water is pumped into the ECP, increasing level. This additional water is required, along with that maintained in the ECP, to ensure a 64.5 inch depth, which corresponds to a 30 day supply of cooling water.

BASES (continued)

ACTIONS

~~A.1~~

~~If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days.~~

~~The 7 day Completion Time is reasonable, based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable in one or more cooling towers, the number of available systems, and the time required to complete the Required Action.~~ 19

A.1 and A.2

If ~~the cooling tower fan cannot be restored to OPERABLE status within the associated Completion Time, or~~ if the ECP UHS is inoperable ~~[for reasons other than Condition A]~~, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 19

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (together with SR 3.7.8.3)

This SR verifies that adequate long term (30 days) cooling inventory is available. ~~can be maintained~~. The level specified also ensures NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the ECP level ~~parameter variations~~ during the applicable MODES. This SR verifies that the UHS water level is ≥ 5 ~~[ ]~~ ft ECP indicated ~~[mean sea level]~~. 20

SR 3.7.8.2

This SR provides assurance ~~verifies~~ that the heat sink for the SWS can dissipate ~~cool the CCW System to at least its maximum design temperature within the maximum~~ 20

(continued)

.SES

<INSERT B3.7-49A> 20

JRVEILLANCE
:QUIREMENTS

SR 3.7.~~9~~ 8.2 (continued)

accident or normal heat loads for 30 days following ~~a~~ the Design Basis ~~Accident~~ event. The 24 hour Frequency is based on operating experience related to the trending of the ~~parameter~~ ECP temperature ~~variations~~ during the applicable MODES. This SR verifies that the ~~UHS~~ ECP average water temperature at the point of discharge from the emergency cooling pond (i.e., SWS suction) is ≤ ~~[90]~~ 100 °F. 20

<INSERT B3.7-49B>

SR 3.7.~~9~~ 8.3

<INSERT B3.7-49C>

~~Operating each cooling tower fan for ≥ [15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.~~ 19 20

REFERENCES

1. ~~F~~SAR, Section ~~[9.2.5]~~ 9.3 edit
2. Regulatory Guide 1.27, Rev. 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974. edit 6
3. 10 CFR 50.36.

<INSERT B3.7-49A>

The measured temperature at the discharge from the ECP is considered a conservative average of total ECP conditions since solar gain, wind speed, and thermal current effects throughout the ECP will essentially be at equilibrium conditions under initial stagnant conditions.

<INSERT B3.7-49B>

This SR is modified by a Note indicating that the temperature monitoring is required to be performed only during the summer months (i.e., June 1 to September 30). During other periods of the year, the ECP temperature will not have the potential to reach the temperature limit.

<INSERT B3.7-49C>

This SR (together with SR 3.7.8.1) verifies that adequate inventory exists to support long term (30 days) cooling. The volume specified is verified by soundings to confirm that the indicated level is consistent with the assumed level, and by visual inspections of the physical properties of the ECP. An engineering evaluation is performed of any apparent changes in visual appearance or other abnormal degradation to verify the capability of the ECP to fulfill its safety function. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

B 3.7 PLANT SYSTEMS

B 3.7.~~10~~ 9 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS is a shared system which provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity~~, chemicals, or toxic gas~~.

The CREVS consists of two independent, ~~a~~ redundant, and fan filter assemblies. Each filter train consists of a roughing filter, a high efficiency particulate air (HEPA) filter, and a charcoal ~~filter~~ adsorber. fan circulates control room air through a for control room pressurization each train provides additional outside air filtered through a four inch bed of charcoal adsorbers

~~The CREVS is an emergency system.~~ Upon receipt of ~~the~~ a unit specific high radiation ~~activating~~ signal~~(s)~~, the associated unit's normal control room ventilation system is ~~automatically~~ shut down, the control room envelope is isolated, and the CREVS ~~can be~~ is ~~manually~~ started. ~~The roughing filters and water condensing units remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA and charcoal filters.~~ The control room envelope is maintained sufficiently leak tight to minimize unfiltered air inleakage. ~~A single train will pressurize the control room with a 1.5 ft² LEAKAGE area to about 1/8 inch water gauge.~~ The CREVS operation is discussed in the ~~F~~ UFSAR, Section ~~9.4~~ 9.7 (Ref. 1).

The CREVS is designed to maintain the control room for 30 days of continuous occupancy after a Design Basis Accident (DBA), without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The shared CREVS components are arranged in ~~redundant~~ two safety related ventilation trains. ~~The location of components and ducting within the control room envelope~~ which ensure~~s~~ an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators ~~as demonstrated by the control room accident dose analyses~~ for the ~~most limiting~~ design basis loss of coolant accident fission product release ~~presented in the FSAR, Chapter [15] (Ref. 2)~~ and for a fuel handling accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case single active failure of a CREVS component, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

~~For this unit, there are no sources of toxic gases or chemicals that could be released to affect control room habitability.~~ 39

In MODES 1 and 2, and during movement of irradiated fuel assemblies, the CREVS satisfies Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 2). 6

<INSERT B3.7-51B>

edit

LCO

Two ~~independent and redundant~~ CREVS trains are required to be OPERABLE to ensure that at least one is available if a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

edit

For a ~~The~~ CREVS train to be ~~is~~ considered OPERABLE, ~~when~~ the ~~individual components necessary to control operator exposure are OPERABLE in both trains. A~~ CREVS train ~~is considered OPERABLE when~~ the associated must include: 39

<INSERT B3.7-51A>

a. ~~Fan is OPERABLE;~~

b. OPERABLE HEPA filter and charcoal adsorber ~~are not excessively restricting flow, and are capable of performing their filtration functions~~; and

c. ~~Heater, demister,~~ ductwork, ~~valves,~~ and dampers ~~are OPERABLE, and~~ air circulation can be ~~maintained~~ sufficient to maintain. 39

In addition, the control room ~~boundary~~ envelope, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis. and provide adequate makeup air flow. 32

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the control room will remain habitable during and following a DBA.

(continued)

<INSERT B3.7-51A>

OPERABLE fan capable of being powered from both a normal and an OPERABLE emergency power source (Note: Because this is a shared system and may be powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREVS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.9 must be applied for inoperable CREVS train(s).);

<INSERT B3.7-51B>

In MODES 3 and 4, the CREVS satisfies Criterion 4 of 10 CFR 50.36.

BASES

APPLICABILITY (continued)

During movement of irradiated fuel assemblies ~~[and during CORE ALTERATIONS]~~, the CREVS must be OPERABLE to cope with a release due to a fuel handling accident.

ACTIONS

A.1

With one CREVS train inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room radiation protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 ~~[, C.2.1,]~~ and ~~[C.2.2]~~

~~[In MODE 5 or 6, or]~~ during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~, if the ~~inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time,~~ Required Action and associated Completion Time of Condition A are not met the OPERABLE CREVS train must immediately be placed in the emergency recirculation mode. This action ensures that ~~the remaining train is OPERABLE, that~~ no failures preventing automatic actuation will occur, and that any active failure will be readily detected. ~~Required Action C.1 is modified by a Note indicating to place the system in the emergency mode if automatic transfer to emergency mode is inoperable.~~

9 edit

39

37

(continued)

BASES

ACTIONS

C.1, C.2, ~~and~~ ~~C.2.2~~ (continued)

An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

movement of irradiated fuel assemblies since this is an activity

edit

D.1

[~~In MODE 5 or 6, or~~] during movement of irradiated fuel assemblies [~~, or during CORE ALTERATIONS~~], when two CREVS trains are inoperable, action must be taken immediately to suspend ~~activities~~ that could release radioactivity that could enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

edit

E.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, the CREVS may not be capable of performing the intended function and ~~the unit is in a condition outside the accident analysis.~~ a loss of safety function has occurred. Therefore, LCO 3.0.3 must be entered immediately.

edit

SURVEILLANCE REQUIREMENTS

SR 3.7.~~10~~ 9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks this system. ~~Monthly heater operations dry out any moisture that has accumulated in the charcoal because of humidity in the ambient air. [Systems with heaters must be operated for > 10 continuous hours with the heaters energized. Systems without heaters~~ need only be operated for ≥ 15 minutes to demonstrate the function of the system. ~~]~~ The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

39

and

This test is conducted on alternating trains semimonthly by initiating flow through the roughing filters, HEPA filters and charcoal adsorbers. The CREVS is designed

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.~~10~~ 9.2

This SR verifies that the required CREVS testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. ~~The CREVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3).~~ The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal a~~b~~dsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~. 39 edit

SR 3.7.~~10~~ 9.3

This SR verifies that ~~[~~each CREVS train starts ~~[or the control room isolates]~~ and operates within 10 seconds, and with flow through the HEPA filters and charcoal adsorber banks on an actual or simulated actuation signal. The Frequency of ~~[~~18~~]~~ months is consistent with ~~that specified~~ the guidance provided in ~~Reference 3~~ Regulatory Guide 1.52 (Ref. 3). edit edit

~~SR 3.7.10.4~~

~~This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify that the CREVS is functioning properly. During the emergency mode of operation, the CREVS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure with respect to adjacent areas, to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train at a flow rate of ≤ [3300] cfm. This value includes [300] cfm of outside air. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.~~ 31

REFERENCES

1. ~~F~~SAR, Section ~~[9.4.3]~~ 9.7. edit 6
2. ~~FSAR, Chapter [15]~~ 10 CFR 50.36. edit
3. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

B 3.7 PLANT SYSTEMS

B 3.7.~~14~~ 10 Control Room Emergency Air Conditioning ~~Temperature Control~~ System (CREA~~T~~CS)

BASES

BACKGROUND

The CREA~~T~~CS provides temperature control for the control room following isolation of the control room.

The CREA~~T~~CS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, ~~valves or~~ dampers, and instrumentation also form part of the system. ~~Two redundant air cooled condensing units are provided as a backup to the water cooled condensing unit. Both the water cooled and air cooled condensing units must be OPERABLE for the CREATCS to be OPERABLE.~~ During ~~emergency~~ operation, the CREA~~T~~CS maintains the temperature in a range consistent with personnel comfort and long term equipment operation. ~~between 70°F and 85°F.~~ The CREA~~T~~CS is a subsystem ~~providing air temperature control for the control room.~~ edit

~~The CREATCS is an emergency system.~~ On detection of high ~~containment building pressure or~~ radiation, ~~low Reactor Coolant System pressure, or high noble gas radioactivity in the station vent,~~ the control room envelope is isolated, the normal control room ventilation system is ~~automatically~~ shut down, and the CREACS is ~~Control Room Emergency Ventilation System can be manually~~ started. A single train will provide the required temperature control. The CREA~~T~~CS operation to maintain control room temperature is discussed in the ~~F~~SAR, Section ~~[9.4]~~ 9.7 (Ref. 1).

40

APPLICABLE SAFETY ANALYSES

The design basis of the CREA~~T~~CS is to maintain control room temperature for 30 days of continuous occupancy.

The CREA~~T~~CS components are arranged in redundant, safety related trains. ~~During emergency operation, the CREATCS maintains the temperature between [70]°F and [95]°F.~~ A single active failure of a CREA~~T~~CS component does not impair the ability of the system to perform as designed. The CREA~~T~~CS is designed in accordance with Seismic Category I requirements. The CREA~~T~~CS is capable of removing sensible and latent heat loads from the control room, including

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

consideration of equipment heat loads and personnel occupancy requirements, to ensure ~~equipment OPERABILITY~~ a habitable environment and

In MODES 1 and 2, and during movement ~~of irradiated fuel assemblies,~~ The CREAVCS satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2). <INSERT B3.3-56B>

LCO

Two independent and redundant trains of the CREAVCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other ~~equipment~~ train. Total system failure could result in the ~~operating~~ control room temperature exceeding limits in the event of an accident. <INSERT B3.7-56A>

For a ~~The~~ CREAVCS train to be ~~is~~ considered OPERABLE, ~~when~~ the individual components that are necessary to maintain control room temperature must be ~~are~~ OPERABLE ~~in both trains~~. These components include the cooling coils, ~~water cooled~~ condensing units, and associated temperature control instrumentation. In addition, the CREAVCS must be ~~OPERABLE to the extent that~~ capable of maintaining air circulation ~~can be maintained~~.

APPLICABILITY

In MODES 1, 2, and 3, ~~4, [5, and 6,]~~ and during movement of irradiated fuel assemblies ~~and during CORE ALTERATIONS~~, the CREAVCS must be OPERABLE to ensure that the control room temperature will not exceed ~~equipment OPERABILITY requirements~~ habitability and equipment OPERABILITY requirements following isolation of the control room.

ACTIONS

A.1

With one CREAVCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREAVCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CREAVCS train could result in a loss of CREAVCS function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining train can provide the required capabilities, and ~~the~~ alternate ~~safety or~~ nonsafety related cooling means that are available.

(continued)

<INSERT B3.7-56A>

(Note: Because this is a shared system and is normally powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREVS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.10 must be applied for inoperable CREACS train(s).)

<INSERT B3.7-56B>

In MODES 3 and 4, the CREACS satisfies Criterion 4 of 10 CFR 50.36.

BASES

ACTIONS

A.1 (continued)

~~Concurrent failure of two CREATCS trains would result in the loss of function capability; therefore, LCO 3.0.3 must be entered immediately.~~

edit

B.1 and B.2

In MODE 1, 2, 3, or 4, if the ~~inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time,~~ Required Action and associated Completion Time of Condition A are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

40

C.1 and C.2

~~[In MODE 5 or 6, or]~~ ~~d~~During movement of irradiated fuel ~~[, or during CORE ALTERATIONS]~~, if the ~~inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time~~, the OPERABLE CREA~~T~~CS train must be placed in operation immediately. This action ensures that ~~the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that~~ any active failure will be readily detected.

40

An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require the isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

~~[In MODE 5 or 6, or]~~ ~~d~~During movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~, with two CREA~~T~~CS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places

(continued)

BASES

ACTIONS

D.1 (continued)

the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREA~~T~~CS trains are inoperable in MODE 1, 2, 3, or 4, ~~the CREATCS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore,~~ a loss of safety function has occurred, and LCO 3.0.3 must be entered immediately.

edit

edit

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

<INSERT B 3.7-58A>

~~This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses]. This SR consists of a combination of testing and calculations. An [18] month Frequency is appropriate, as significant degradation of the CREATCS is slow and is not expected over this time period.~~ 52

REFERENCES

1. ~~F~~SAR, Section ~~[9.4]~~ 9.7.
2. 10 CFR 50.36.

edit 6

<INSERT B3.7-58A>

SR 3.7.10.1 and SR 3.7.10.2

These SRs, in conjunction with periodic preventative maintenance activities, provide verification that the CREACS will maintain the control room temperature within acceptable bounds. SR 3.7.10.1 is performed on a staggered basis with one train being tested every two weeks. The Frequencies (31 days and 18 months) are appropriate as periodic preventative maintenance activities are routinely performed and significant degradation of the CREACS is not expected over these time periods.

B 3.7 PLANT SYSTEMS

B 3.7.~~12~~ 11 [redacted] Penetration Room Ventilation System (~~E~~PR VS)

BASES

BACKGROUND

The ~~E~~PRVS filters air from the ~~area of the active Emergency Core Cooling System (ECCS) components during the recirculation phase of~~ penetration areas in the event of penetration leakage from the reactor building ~~a loss of coolant accident (LOCA)~~.

The ~~E~~PRVS consists of two independent, redundant trains. Each train consists of ~~a~~ prefilter, a high efficiency particulate air (HEPA) filter and ~~an~~ activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, ~~valves~~ or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the ~~Auxiliary Building negative pressure area~~ penetration rooms following receipt of ~~a safety features~~ an engineered safeguards actuation system (ESAS) signal. ~~(SFAS)~~

<INSERT from ASA discussion B 3.7-60>

~~The EVS is a standby system. During emergency operations, the EVS dampers are realigned, and fans are started to begin filtration.~~ Upon receipt of the ~~SFAS~~ ESAS signal(s), normal air discharges from the ~~negative pressure~~ area are isolated, and ~~the stream of ventilation~~ air discharged through the system filter ~~trains~~ (s). The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. 41

The ~~E~~PRVS is discussed in the ~~F~~SAR, Sections ~~[6.2.3], [9.4.2]~~ 6.5 and ~~[15.4.6]~~ 14.2.2.5 (Refs. 1, ~~2~~, and ~~3~~ 2, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the ~~E~~PRVS is established by the ~~large break~~ LOCA. The system ~~evaluation assumes a passive failure of the ECCS outside containment, such as an ECCS pump seal failure during the recirculation mode. In such a case, the system limits radioactive release to within 10 CFR 100 (Ref. 4) requirements.~~ provides filtration for the most likely location of reactor building leakage, i.e., at the penetrations. The analysis of the effects and consequences of a ~~large break~~ LOCA is presented in Reference ~~3~~ 2. ~~The EVS also actuates following a small break LOCA, in those cases where the unit goes into the recirculation mode of long term cooling, and to cleanup releases of smaller leaks, such as from valve stem packing.~~ 41

(continued)

BASES

edit

APPLICABLE SAFETY ANALYSES (continued)

~~Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal of any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.~~

Insert in Background

Following a LOCA, an ESFAS signal starts the lead PRVS ~~EVS fans~~ and ~~opens the dampers located in the penetration room outlet ductwork. The ESFAS signal closes all containment isolation valves and purge system valves. The purge system fans, if running, are shut down automatically.~~

If proper flow is not achieved within 20 seconds, the lead system is automatically stopped and 5 seconds later the standby system starts.

41

In MODES 1 and 2, the PRVS satisfies Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 3).

6

LCO

<INSERT B3.7-60A>

Two ~~independent and~~ redundant trains of the PRVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. ~~Total system failure could result in atmospheric release from the negative pressure area boundary exceeding Reference 4 limits in the event of a Design Basis Accident (DBA).~~

41

edit

~~The EVS is considered OPERABLE when the individual components necessary to maintain the negative pressure area boundary filtration are OPERABLE in both trains.~~

For a ~~An~~ PRVS train ~~is~~ to be considered OPERABLE, ~~when~~ its associated:

a. Fan ~~is~~ must be OPERABLE;

b. HEPA filter and charcoal adsorber ~~are~~ must be not excessively restricting flow, and ~~are~~ must be capable of performing their filtration functions; and

c. ~~Heater, demister,~~ Required ductwork, ~~valves~~ and dampers ~~are~~ must be OPERABLE. ~~and air circulation can be maintained.~~

41

APPLICABILITY

In MODES 1, 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the OPERABILITY requirements of the ~~ECCS~~ reactor building.

In MODES 5 and 6, the PRVS is not required to be OPERABLE since the ~~ECCS~~ reactor building is not required to be OPERABLE.

41

(continued)

<INSERT B3.7-60A>

In MODES 3 and 4, the PRVS satisfies Criterion 4 of 10 CFR 50.36.

BASES (continued)

ACTIONS

A.1

With one ~~E~~ PR VS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ~~E~~VS PR safety function. However, the overall reliability is reduced because a single failure in the OPERABLE ~~E~~VS train could result in loss of ~~E~~VS PR function.

The 7 day Completion Time is appropriate reactor building because the risk contribution is less than that of the ~~ECCS~~ (~~72~~ 1 hour Completion Time), and this system is not a direct support system for the ~~ECCS~~ reactor building. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

41

B.1 and B.2

If the Required Action and ~~EVS train cannot be restored to OPERABLE status within~~ the associated Completion Time are not met, or with both PRVS trains inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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50

SURVEILLANCE REQUIREMENTS

SR 3.7.~~12~~ 11.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. ~~Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.]~~ The 31 day Frequency is based on known reliability of equipment and the two train redundancy available.

41

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2 11

This SR verifies that the required EVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. ~~The EVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5).~~ The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal ~~(general use and following specific operations)~~. Specific test frequencies and additional information are discussed in detail in the [VFTP].

41

SR 3.7.12.3 11

This SR verifies that each EVS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with ~~that specified~~ the guidance provided in ~~Reference 5~~ Regulatory Guide 1.52 (Ref. 4).

edit

~~SR 3.7.12.4~~

~~This SR verifies the integrity of the negative pressure boundary area. The ability of the EVS to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the EVS. During the [post accident] mode of operation, the EVS is designed to maintain a slight negative pressure in the negative pressure boundary area with respect to adjacent areas to prevent unfiltered LEAKAGE. The EVS is designed to maintain this negative pressure at a flow rate of [3000] cfm from the negative pressure boundary area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.~~

36

~~SR 3.7.12.5~~

~~Operating the EVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the EVS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference 5.~~

33

(continued)

BASES (continued)

REFERENCES

1. ~~F~~USAR, Section ~~[6.2.3]~~. 6.5. edit

2. ~~F~~USAR, Sections ~~[9.4.2]~~. 14.2.2.5 and 14.2.2.6 edit

~~3. FSAR, Section [15.4.6].~~ edit 6

~~4. 10 CFR 100.11.~~ edit

3. 10 CFR 50.36.

4 ~~5.~~ Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978. edit

B 3.7 PLANT SYSTEMS

B 3.7.~~13~~ 12 Fuel ~~Storage Pool~~ Handling Area Ventilation System ~~(FSPVS)~~ (FHAVS)

BASES

BACKGROUND

The ~~FSPVS~~ FHAVS ~~provides negative pressure in the fuel storage area, and~~ filters airborne radioactive ~~particulates~~ from the area of the spent fuel pool following a fuel handling accident. material

edit

The ~~FSPVS~~ FHAVS consists of portions of the normal ~~Fuel Handling Area Ventilation System (FHAVS), the station Emergency Ventilation System (EVS) ductwork bypasses, and dampers. The portion of the normal FHAVS used by the FSPVS consists of ducting between the spent fuel pool and the normal FHAVS exhaust fans or dampers, and redundant radiation detectors installed close to the suction end of the FHAVS exhaust fan ducting. The portion of the EVS used by the FSPVS consists of two independent, redundant trains. Each train consists of a~~ Auxiliary Building Heating, Ventilation, and Air Conditioning System. The FHAVS single train which includes a supply fan, ~~heater,~~ prefilter, ~~a~~ high efficiency particulate air (HEPA) filter, activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two exhaust fans. Ductwork, ~~valves or~~ dampers, and instrumentation also form part of the system. ~~Two isolation valves are installed in series in the ductwork between the FHAVS and the EVS to provide isolation of the EVS from the FHAVS on an Engineered Safety Feature actuation signal. These valves are opened prior to fuel handling operations. The EVS is the subject of LCO 3.7.12, "Emergency Ventilation System (EVS)," and is fully described in the FSAR, Section [6.2.3], Reference 12. A ductwork bypass with redundant dampers connects the FHAVS to the EVS.~~

42

During ~~normal~~ operation, the exhaust from the fuel handling area is passed through the FHAVS exhaust filter and is discharged through the station vent stack. ~~In the event of a fuel handling accident, the radiation detectors (one per EVS train), located at the suction of the FHAVS exhaust fan ducting, send signals to isolate the FHAVS supply and exhaust fans and ductwork, open the redundant dampers in the bypass ductwork, and start the EVS fans. The EVS fans pull the air from the fuel handling area, creating a negative pressure, and discharge the filtered air to the station vent.~~

The FHAVS is discussed in the ~~F~~USAR, Sections ~~[6.2.3]~~ 9.7, ~~[9.4.2]~~, and ~~[15.4.7]~~ 14.2.2 (Refs. 1, ~~2~~, and ~~3~~ 2, respectively).

edit

(continued)

BASES

BACKGROUND (continued)

~~because it may be used for normal as well as post accident, atmospheric cleanup functions.~~ 42

APPLICABLE SAFETY ANALYSES

The FHAVS ~~FSPVS~~ design basis is established by the the amount of ~~consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3 2, assumes that a certain number of fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FSPVS is functional due to a single failure that disables the other train. The accident analysis accounts for the~~ credits the FHAVS for a reduction in airborne radioactive material released to the environment. ~~provided by the remaining one train of this filtration system.~~ These assumptions and the analysis ~~follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).~~ are further discussed in Reference 2. 42

The FHAVS ~~FSPVS~~ satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 6

LCO

During movement of irradiated fuel, ~~[Two] independent and redundant trains of~~ the FHAVS is ~~FSPVS are~~ required to be OPERABLE and operating to ensure that ~~at least one is available, assuming a single failure that disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel handling area exceeding 10 CFR 100 (Ref. 5) limits in the event of a fuel handling accident.~~ 35 42

~~The FSPVS is considered OPERABLE when the individual components necessary to control operator exposure in the fuel handling building are OPERABLE in both trains. An~~ For The FHAVS ~~FSPVS train is~~ considered OPERABLE: ~~when its associated:~~ to be

1. One exhaust fan must be ~~Fan is~~ OPERABLE;
2. HEPA filter and charcoal adsorber must ~~are~~ not be excessively restricting flow, and must be ~~are~~ capable of performing their filtration functions; and
3. ~~[Heater, demister,]~~ ductwork, valves, and dampers must be ~~are~~ OPERABLE. ~~and air circulation can be maintained~~

The FHAVS must be operating since it does not automatically start following a fuel handling accident. A supply fan may be operating, but is not required for FHAVS OPERABILITY. 35

(continued)

BASES (continued)

APPLICABILITY

~~In [MODES 1, 2, 3, and 4,] the FSPVS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a loss of coolant accident (refer to LCO 3.7.12) for units that use this system as part of their EVSs.~~ 34

During movement of FHAVS irradiated fuel assemblies in the fuel handling area, the ~~FSPVS~~ is always required to be OPERABLE and operating to mitigate the consequences of a fuel handling accident. 35

~~In MODES 5 and 6, the FSPVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.~~ 34

(INSERT B3.7-66A) 34

ACTIONS

A.1

~~With one FSPVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time period, the remaining OPERABLE train is adequate to perform the FSPVS function. However, the overall reliability is reduced because a single failure in the OPERABLE FSPVS train could result in a loss of FSPVS functioning. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable FSPVS train, and ability of the remaining FSPVS train to provide the required protection.~~ 34

~~B.1 and B.2~~

~~In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the associated Completion Time, or when both FSPVS trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

~~C.1 and C.2~~

~~If the inoperable FSPVS train cannot be restored to OPERABLE status within the required Completion Time, during movement~~

(continued)

<INSERT B3.7-66A>

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note which states that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

ACTIONS

~~C.1 and C.2 (continued)~~

~~of irradiated fuel assemblies in the fuel handling area, the OPERABLE FSPVS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failures will be readily detected.~~

~~If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This action does not preclude the movement of fuel assemblies to a safe position.~~

34

~~D.1~~

35

When ~~two trains of~~ the ~~FSPVS are~~ FHAVS is inoperable or not in operation during movement of irradiated fuel assemblies in the fuel handling area, ~~the unit must be placed in a condition in which the LCO does not apply. This LCO involves~~ immediately suspending movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

INSERT B3.7-67A

edit

35

INSERT B3.7-67B

SURVEILLANCE REQUIREMENTS

~~SR 3.7.13.1~~

~~Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.~~

SR 3.7.~~13~~ 12.2

This SR verifies that the required ~~FSPVS~~ FHAVS testing is performed in accordance with the ~~[~~Ventilation Filter Testing

(continued)

<INSERT B3.7-67A>

immediate action must be taken to preclude the occurrence of an accident. This is achieved by

<INSERT B3.7-67B>

SR 3.7.12.1

Periodic verification of the operation of the FHAVS assures immediate availability of filtration following a fuel handling accident. A 12 hour Frequency is sufficient, considering the system indications and alarms available to the operator for monitoring the FHAVS in the control room.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~13~~ 12.2 (continued)

Program (VFTP)~~]~~ 9. ~~The FSPVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6).~~ The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal ~~(general use and following specific operations)~~. Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.

42

edit

~~SR 3.7.13.3~~

~~This SR verifies that each FSPVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with that specified in Reference 6.~~

35

~~SR 3.7.13.4~~

~~This SR verifies the integrity of the fuel handling area. The ability of the fuel handling area to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the FSPVS. During the [post accident] mode of operation, the FSPVS is designed to maintain a slight negative pressure in the fuel handling area to prevent unfiltered LEAKAGE. The FSPVS is designed to maintain this negative pressure at a flow rate of ≤ [3000] cfm to the fuel handling area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice.~~

36

~~SR 3.7.13.5~~

~~Operating the FSPVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FSPVS filter bypass damper is verified if it can be opened. A Frequency of [18] months is specified in Reference 6.~~

33

(continued)

BASES (continued)

REFERENCES

1. ~~F~~USAR, Section ~~6.2.3~~. 9.7.

~~2. FSAR, Section 9.4.2.~~

2 ~~3~~. ~~F~~USAR, Section ~~15.4.7~~. 14.2.2.

3. 10 CFR 50.36.

~~4. Regulatory Guide 1.25.~~

~~5. 10 CFR 100.11.~~

~~6. Regulatory Guide 1.52.~~

edit

6

edit

10

B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumption of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Section [9.1.2], Reference 1. The Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section [9.1.3] (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section [15.4.7] (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is below 10 CFR 100 (Ref. 5) guidelines.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With 23 ft, the assumptions of Reference 4 can be used directly. In practice, the LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel rack, however, there may be < 23 ft above the top of the fuel bundle and the surface, by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although the analysis shows that only the first [few] rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the fuel storage pool at less than the required level, the movement of fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of loads over the spent fuel. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level

(continued)

(10)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.14.1 (continued) changes are controlled by unit procedures and are acceptable, based on operating experience. During refueling operations, the level in the fuel storage pool is at equilibrium with that in the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.
REFERENCES	1. FSAR, Section [9.1.2]. 2. FSAR, Section [9.1.3]. 3. FSAR, Section [15.4.7]. 4. Regulatory Guide 1.25. 5. 10 CFR 100.11.

44

B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ 13 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the ~~following~~ Bases for LCO 3.7.~~15~~ 14, "Spent Fuel ~~Assembly~~ Pool Storage," fuel assemblies are stored in the spent fuel pool racks ~~[in a "checkerboard" pattern]~~ in accordance with criteria based on ~~initial~~ enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ ~~[1500]~~ 1600 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron.

edit

INSERT B3.7-73A

APPLICABLE SAFETY ANALYSES

INSERT B3.7-73B

~~A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.16 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the extreme case of completely loading the spent fuel pool racks with unirradiated assemblies of maximum enrichment. Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded spent fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios.~~

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 4). 6

LCO

The specified concentration ~~[≤ 1500]~~ ≥ 1600 ppm of dissolved boron in the spent fuel ~~storage~~ pool conservatively preserves the assumption used in the analyses of the potential accident scenarios ~~described above~~. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel ~~storage~~ pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool

(continued)

<INSERT B3.7-73A>

in the spent fuel pool water.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.14-1. Fuel assemblies not meeting the criteria of Figure 3.7.14-1 shall be stored in accordance with paragraph 4.3.1.1.e in SAR Section 4.3, Fuel Storage.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Thus, for accident conditions, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic condition. For example, accident scenarios are postulated which could potentially increase the reactivity and reduce the margin to criticality. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the high density storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.14, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.14.1.

<INSERT B3.7-73B>

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more that eight inches of water separating it from the active fuel in the rest of the rack which precludes interaction). However, accidents can be postulated which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Thus, for accident conditions, the presence of soluble boron in the storage pool water is assumed as a realistic initial condition.

The presence of 1600 ppm boron in the pool water will decrease reactivity by approximately 30% ΔK. Thus $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

BASES

APPLICABILITY (continued)

verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movement in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the Spent fuel ~~storage~~ pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position.

INSERT B3.7-74A

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

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SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that the concentration of boron in the spent fuel ~~storage~~ pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

edit

REFERENCES

~~None.~~ INSERT B3.7-74B

<INSERT B3.7-74A>

In addition, action must be immediately initiated to restore the spent fuel pool boron concentration to within its limit. An acceptable alternative is to immediately initiate performance of a spent fuel pool verification to ensure proper locations of the fuel since the last movement of fuel assemblies in the spent fuel pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. Either of these actions are acceptable, and once initiated must be continued until the action is completed. The immediate Completion Time for initiation of these actions reflects the importance of maintaining a controlled environment for irradiated fuel.

<INSERT B3.7-74B>

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. SAR, Section 14.2.2.3.
3. Safety Evaluation Report, Section 2.1.3, License Amendment No. 76, April 15, 1983.
4. 10 CFR 50.36.

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B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ 14 Spent Fuel ~~Assembly~~ Pool Storage

BASES

BACKGROUND

The spent fuel assembly storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The ~~storage~~ spent fuel pool is sized to store ~~[735]~~ 968 ~~irradiated~~ fuel assemblies. ~~which includes storage for [15] failed fuel containers.~~ The spent fuel storage cells are installed in parallel rows with center to center spacing of ~~[12 31/32]~~ 10.65 inches in ~~one direction, and [13 3/16] inches in the other orthogonal~~ each direction. ~~This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans,~~ <INSERT B3.7-75A> is sufficient to maintain a k_{eff} of < 0.95 for spent fuel of original enrichment of up to ~~[3.5]~~ 4.10%. However, ~~as higher initial enrichment~~ Region 2 fuel assemblies ~~are~~ which do not meet enrichment and burnup criterion ~~are~~ stored in the spent fuel pool. ~~they~~ must be stored in a checkerboard pattern ~~taking into account fuel burnup~~ to maintain a k_{eff} of 0.95 or less. <INSERT B3.7-75B>

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APPLICABLE SAFETY ANALYSES

~~The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.~~ <INSERT B3.7-75C>

The spent fuel ~~assembly~~ pool storage satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3).

edit 49 · 6

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to Figure ~~[3.7.16-1]~~ 3.7.14-1 ~~in the accompanying LCO,~~ ensure that the k_{eff} of the spent fuel pool will always remain ≤ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent ~~[ ]~~ fuel pool. ~~according to Figure [3.7.16-1]~~ Fuel assemblies not meeting the enrichment and burnup criteria ~~of Figure [3.7.16-1]~~ shall be stored in accordance with Specification 4.3.1.1.

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(continued)

<INSERT B3.7-75A>

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.14-1. Fuel assemblies not meeting the criteria of Figure 3.7.14-1 shall be stored in accordance with paragraph 4.3.1.1.e in SAR Section 4.3, Fuel Storage.

<INSERT B3.7-75B>

In order to prevent inadvertent fuel assembly insertion into two adjacent storage locations, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (unrestricted) are physically blocked before any such fuel assembly is placed in Region 2 (Ref. 1). In addition, the area designated for checkerboard arrangement is divided from the normal storage in Region 2 by a row of vacant storage spaces (Ref. 2).

<INSERT B3.7-75C>

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies in Region 1. Region 2 controls fuel assembly interaction by fixing the minimum separation between assemblies and by setting enrichment and burnup criterion to limit fissile materials. This

BASES (continued)

APPLICABILITY	This LCO applies whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure [3.7.~~16~~ 14-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure [3.7.~~16~~ 14-1] or Specification 4.3.1.1. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.~~16~~ 14.1 This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.~~16~~ 14-1] in the accompanying LCO or Specification 4.3.1.1. For fuel assemblies in the unacceptable range of Figure 3.7.14-1, performance of the SR will ensure compliance with Specification 4.3.1.1.
REFERENCES	~~None.~~ 1. SAR, Section 9.6.2. 2. Safety Evaluation Report for ANO-1 License Amendment No. 76, Section 2.1 (OCNA 048314) dated April 15, 1983. 3. 10 CFR 50.36.

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B 3.7 PLANT SYSTEMS

B 3.7.~~17~~ 4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, ~~anticipated operational occurrences~~ abnormalities, and accidents. 51

12 This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational Leakage") of primary coolant at the limit of 3.5 ~~1.0~~ μCi/gm (LCO 3.4.~~16~~, "RCS Specific Activity"). ~~The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant leakage. Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.~~ 44

~~With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.79 rem if the main steam safety valves (MSSVs) are open for the 2 hours following a trip from full power.~~ INSERT B 3.7-77A 24

Operating a unit at the allowable limits could result in a 2 hour ~~EAB~~ exclusion area boundary (EAB) exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits~~, or the limits established as the NRC staff approved licensing basis~~. 44

(continued)

<u><INSERT B3.7-77A></u>

The thyroid dose conversion factors used in the calculation of DOSE EQUIVALENT I-131 are those identified in Section 1.1, "Definitions."

BASES (continued)

APPLICABLE SAFETY ANALYSES

INSERT B3.7-78A

~~The accident analysis of the main steam line break, as discussed in the FSAR, Chapter [15] (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed established limits, (Ref. 1) for whole body and thyroid dose rates.~~ 44

~~With a loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Emergency Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure has decreased sufficiently for the Shutdown Cooling System to complete the cooldown.~~

~~In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.~~

In MODES 1 and 2, ~~Secondary~~ secondary specific activity limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 6

In MODES 3 and 4, the secondary specific activity limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ ~~[0.10]~~ 0.17 μCi/gm DOSE EQUIVALENT I-131 maintains the radiological consequences of a Design Basis Accident (DBA) ~~to a small fraction of~~ significantly less than the Reference 1 ~~limits~~ guideline doses. 44

~~Monitoring the specific activity of the secondary coolant ensures that, when secondary specific activity limits are exceeded, appropriate actions are taken, in a timely manner,~~

(continued)

<INSERT B3.7-78A>

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside the reactor building and a loss of load incident were considered (Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975 (Ref. 2)).

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released (Ref. 2).

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for LCO 3.4.13 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for LCO 3.4.13. For the less probable accident of a steam line break, the assumption is made that a loss of 10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident (Ref. 2).

BASES

LCO (continued)	~~to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.~~
APPLICABILITY	In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere. In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are at low pressure and primary to secondary LEAKAGE is minimal. Therefore, ~~monitoring of~~ secondary specific activity is not ~~required.~~ a concern. edit
ACTIONS	A.1 and A.2 DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.~~17~~4.1 This SR verifies that the secondary specific activity is within the limits of the accident analysis assumptions. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the ~~validity of the safety~~ analysis assumptions ~~as releases.~~ are met. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit. edit edit

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 100.11.
2. ~~FSAR, Chapter [15]~~. Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975. 44
3. 10 CFR 50.36. 6

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B 3.7 PLANT SYSTEMS

B 3.7.18 Steam Generator Level

BASES

BACKGROUND

A principal function of the steam generators is to provide superheated steam at a constant pressure (900 psia) over the power range. Steam generator water inventory is maintained large enough to provide adequate primary to secondary heat transfer. Mass inventory and indicated water level in the steam generator increases with load as the length of the four heat transfer regions within the steam generator vary. Inventory is controlled indirectly as a function of power and maintenance of a constant average primary system temperature by the feedwater controls in the Integrated Control System.

The maximum operating steam generator level is based primarily on preserving the initial condition assumptions for steam generator inventory used in the FSAR steam line break (SLB) analysis (Ref. 1). An inventory of 62,600 lb was used in this analysis. The 62,600 lb must not be exceeded due to the concerns of a possible return to criticality because of primary side cooling following an SLB and the maximum pressure in the reactor building.

For a clean once through steam generator, the mass inventory in a steam generator for operating at 100% power is approximately 39,000 lb to 40,000 lb.

As a steam generator becomes fouled and the operating level approaches the limit of 96%, the mass inventory in the downcomer region increases approximately 10,000 lb, and adds to the total mass inventory of the steam generator. In matching unit data of startup level versus power, the steam generator performance codes have shown that fouling of the lower tube support plates does not significantly change the heat transfer characteristics of the steam generator. Thus, the steam temperature, or superheat, is not degraded due to the fouling of the tube support plates, and mass inventory changes are mainly due to the added level in the downcomer.

Analytically, increasing the fouling of the steam generator tube surfaces degrades the heat transfer capability of the steam generator, increases the mass inventory, and decreases the steam superheat at 100% power (2544 MW). The results

(continued)

BASES

BACKGROUND (continued)

were presented as the amount of mass inventory in each steam generator versus operating range level and steam superheat.

The limiting curve, which was determined from several steam generator performance code runs at a power level of 100%, conservatively bounds steam generator mass inventory value, when operating at power levels < 100%.

The points displayed in Figure 3.7.18-1, in the accompanying LCO, are the intercept points of the 57,000 lb mass value, and the operating range level x and steam superheat values.

The steam generator performance analysis also indicated that startup and full range level instruments are inadequate indicators of steam generator mass inventory at high power levels due to the combination of static and dynamic pressure losses. If the water level should rise above the 96% upper limit, the steam superheat would tend to decrease due to reduced feedwater heating through the aspirator ports. Normally, a reduction in water level is manually initiated to maintain steam flow through the aspirator port by reducing the power level. Thus, the superheat versus level limitation also tends to ensure that, in normal operation, water level will remain clear of the aspirator ports.

Feedwater nozzle flooding would impair feedwater heating, and could result in excessive tube to shell temperature differentials, excessive tubesheet temperature differentials, and large variations in pressurizer level.

APPLICABLE SAFETY ANALYSES

The most limiting Design Basis Accident that would be affected by steam generator operating level is a steam line failure. This accident is evaluated in Reference 1. The parameter of interest is the mass of water, or inventory, contained in the steam generator due to its role in lowering Reactor Coolant System (RCS) temperature (return to criticality concern), and in raising containment pressure during an SLB accident. A higher inventory causes the effects of the accident to be more severe. Figure 3.7.18-1, in the accompanying LCO, is based upon maintaining inventory < 57,000 lb, which is 10% less than the inventory used in the FSAR accident analysis, and therefore is conservative.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The steam generator level satisfies Criterion 2 of the NRC Policy Statement.

LCO

This LCO is required to preserve the initial condition assumptions of the accident analyses. Failure to meet the maximum steam generator level LCO requirements can result in additional mass and energy released to containment, and excessive cooling (and related core reactivity effects) following an SLB. In addition, feedwater nozzle flooding would impair feedwater heating, and could result in excessive tube to shell temperature differentials and excessive tubesheet temperature gradients.

APPLICABILITY

In MODES 1 and 2, a maximum steam generator water level is required to preserve the initial condition assumption for steam generator inventory used in the steam line failure accident analysis (Ref. 1).

In MODE 3, limits on RCS boron concentrations will prevent a return to criticality in the event of an SLB. In MODES 4, 5, and 6, the water in the steam generator has a low specific enthalpy; therefore, there is no need to limit the steam generator inventory when the unit is in this condition.

ACTIONS

A.1

With the steam generator level in excess of the maximum limit, action must be taken to restore the level to within the bounds assumed in the analysis. To achieve this status, the water level is restored to within the limit. The 15 minute Completion Time is considered to be a reasonable time to perform this evolution.

B.1

If the water level in one or more steam generators cannot be restored to less than or equal to the maximum level in

(continued)

(38)

BASES

ACTIONS

B.1 (continued)

Figure 3.7.18-1, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies the steam generator level to be within acceptable limits. The 12 hour Frequency is adequate because the operator will be aware of unit evolutions that can affect the steam generator level between checks. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to steam generator level status.

REFERENCES

1. FSAR, Section [15.4.4].