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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Technical Specification Bases Update to the NRC for the Period April 8, 2003 Through July 23, 2003

Gentlemen:

Pursuant to Waterford Steam Electric Station Unit 3 Technical Specification 6.16, Entergy Operations, Inc. (EOI) hereby submits an update of all changes made to Waterford 3 Technical Specification Bases since the last submittal per letter W3F1-2003-0024, dated April 11, 2003. This TS Bases update is well within the update frequency listed in 10 CFR 50.71(e).

There are no commitments associated with this submittal. Should you have any questions or comments concerning this submittal, please contact Ron Williams at (504) 739-6255.

Very truly yours,

Gautam Sen

G. Sen
Licensing Manager

GS/RLW/ssf

Attachment Waterford 3 Technical Specification Bases Revised Pages

cc: T.P. Gwynn (NRC Region IV), N. Kalyanam (NRC-NRR),
J. Smith, N.S. Reynolds, NRC Resident Inspectors Office

A001

ATTACHMENT 1
TO W3F1-2003-0055

Waterford 3 Technical Specification Bases Revised Pages

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TO W3F1-2003-0055

Waterford 3 Technical Specification Bases Revised Pages

T.S. Bases Change No.	Implement Date	Affected TS Bases Pages	Topic of Change
19	4/23/03	B 3/4 1-1a B 3/4 1-2 B 3/4 4-1 B 3/4 4-1a (new page) B 3/4 8-1a B 3/4 9-1 B 3/4 9-2 B 3/4 9-3	Changes to TS Bases sections were implemented by ER-W3-2003-0131-000 concurrently with TS Amendment 185 that revised TS associated with positive reactivity additions while in shutdown modes. The affected TS sections are: 3.1.2.1, 3.1.2.3, 3.1.2.5, 3.1.2.7, Table 3.3-1, 3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.1.5, 3.8.1.2, 3.8.2.2, 3.8.3.2, 3.9.2, 3.9.8.1, and 3.9.8.2. In addition, the borated water volume requirements in TS 3.1.2.7was changed to be presented in "percent level" units only and an obsolete reference was deleted from Surveillance Requirement 4.8.2.
20	5/5/03	B 2-6	Change to TS Bases section 2.2.1-Reactor Coolant Flow – Low was implemented by ER-W3-2000-0249-000 to correct an oversight related to implementation of TS Amendment 113. Amendment 113 changed the Reactor Coolant Flow – Low trip setpoint in TS Table 2.2-1 from 23.8 psid to 19.00 psid, however, the associated TS Bases section 2.2.1 and FSAR section 7.2.1.1.1.12 were not updated accordingly at that time. This change brings the TS Bases and FSAR into agreement with the currently approved TS.
Reissued 21 & 22	5/7/03	B 3/4 9-1 B 3/4 9-2 B 3/4 9-3 B 3/4 9-4 (new page)	Change No. 21 to TS Bases section 3/4.9.4 was implemented by ER-W3-2002-319-000. Change clarified the TS Bases to be consistent with the TS as amended in TS Amendment 169 to explicitly state the requirements for penetration closure during refueling. Change No. 22 to TS Bases section 3/4.9.9 was implemented by ER-W3-2002-320-000. Change clarified the TS Bases to be consistent with Technical Specifications 3/4.9.4 "Containment Building Penetrations" and 3/4.3.3 "Radiation Monitoring Instrumentation" with respect to operability and closure of the containment purge penetrations during core alterations or movement of irradiated fuel within the containment.
23	5/8/03	B 3/4 0-4a B 3/4 0-5 B 3/4 0-6 B 3/4 0-7 (new page) B 3/4 0-8 (new page)	Changes to TS Bases section 4.0.1 and 4.0.3 were implemented by ER-W3-2002-0407-000 concurrently with TS Amend 187 that revised TS 4.0.1 and 4.0.3 by: (a) standardizing the format and wording of

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Waterford 3 Technical Specification Bases Revised Pages

			general Surveillance Requirements (SR) 4.0.1 and 4.0.3 to be consistent with NUREG-1432, Rev. 2, Improved Standard Technical Specifications (ISTS) wording for SR 3.0.1 and 3.0.3; and (b) modifying the ISTS wording adopted in item (a) above for SR 4.0.3 to extend the delay period allowed to complete a surveillance following the identification of a missed surveillance. In addition, a requirement has been added to SR 4.0.3 to perform a risk evaluation for any surveillance delayed greater than 24 hrs and the risk impact shall be managed.
24	5/28/03	B 3/4 2-2 B 3/4 3-3 B 3/4 7-4a B 3/4 7-5	Changes to various TS Bases sections were implemented by ER-W3-2003-0249-000 concurrently with TS Amendment 188 that made several administrative changes to the Waterford 3 TS e.g. revised, deleted, corrected or clarified certain titles, page numbers, and TS heading information. The amendment also revised personnel and committee titles that have been changed, and revised administrative reporting requirements to conform to 10CFR50.4 and deleted redundant or unnecessary requirements from TS 5.4, 6.6 and 6.7.
25	6/3/03	B 3/4 6-5	Change to TS Bases section 3/4.6.3 was implemented by ER-W3-2003-0257-000. Change revised the TS Bases associated with TS 3.6.3, Containment Isolation Valves, Note 8 of TRM Table 3.6-2 and the TRM Bases associated with TRM 3.6.3. Specifically, the statement "Locked or sealed closed valves may be open on an intermittent basis under administrative control" as contained in the TS and TRM Bases' was changed to clarify that a deactivated automatic valve secured in the isolation position is equivalent to a locked or sealed closed valve and that this statement is applicable to a deactivated automatic valve secured in the isolation position.
26	6/5/03	B 3/4 5-1d	Change to TS Bases section 3/4.5.2 and 3/4.5.3 was implemented by ER-W3-2003-0112-000 to reflect that both LPSI Train A and B have been evaluated for the existence of voids in the system discharge legs. The system design basis has been changed to specify the acceptable volume of voids in each train to be considered operable and full of water.

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27	6/24/03	B 3/4 3-1c B 3/4 3-2	Change to TS Bases section 3/4.3.3.1 was implemented by ER-W3-2002-0235-000 to clarify that it is acceptable to consider the subject radiation monitors operable when the process flow being monitored is intentionally shutdown if the radiation monitors were OPERABLE prior to flow termination.
28	7/23/03	B 3/4 7-8	Change to TS Bases section 3/4.7.12, Essential Services Chilled Water System, was implemented by ER-W3-2000-0991-001 to reflect a change made to the plant to enhance the reliability of the essential chillers.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 19 REPLACEMENT PAGE(S)
(8 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached 8 pages. The revised pages are identified by Change Number 19 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	Insert
B 3/4 1-1a	B 3/4 1-1a
B 3/4 1-2	B 3/4 1-2
B 3/4 4-1	B 3/4 4-1
-----	B 3/4 4-1a
B 3/4 8-1a	B 3/4 8-1a
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The Surveillance Requirements consisting of beginning of cycle measurements, plant parameter monitoring, and end of cycle MTC predictions ensures that the MTC remains within acceptable values. The confirmation that the measured values are within a tolerance of $\pm$ 0.16 X 10^{-4} delta k/k/°F from the corresponding design values prior to 5% power and 40 EFPD provides assurances that the MTC will be maintained within acceptable values throughout each fuel cycle. CE NPSD 911 and CE NPSD 911 Amendment 1, "Analysis of Moderator
Temperature Coefficients in Support of a Change in the Technical Specifications End of Cycle Negative MTC Limit", provide the analysis that established the design margin of $\pm$ 0.16 X 10^{-4} delta k/k/°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 520°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature, and (5) the ECCS analysis remains valid for the peak linear heat rate of Specification 3.2.1.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 2.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions assuming the most reactive CEA stuck out of the core and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and
volumes of Specification 3.1.2.8 plus approximately 19,000 gallons of 2050 ppm borated water from the refueling water storage pool or approximately 58,000 gallons of 2050 ppm borated water from the refueling water storage pool alone. The higher limit of 447,100 gallons is specified to be consistent with Specification 3.5.4 in order to meet the ECCS requirements.

→(DRN 03-375, Ch.19)

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. Temperature changes in the RCS impose reactivity changes by means of the moderator temperature coefficient. Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. This will require a new SDM calculation be performed if the current SDM calculation does not bound the temperature change. Small changes in RCS temperature are unavoidable and so long as the required SDM is maintained during these changes, any positive reactivity additions will be limited to acceptable levels. Introduction of temperature changes must be evaluated to ensure they do not result in a loss of required SDM.

←(DRN 03-375, Ch. 19)

The boron capability required below 200°F is based upon providing a 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5,465 gallons of 2050 ppm borated water from the refueling water storage pool or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops or trains (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

→(DRN 03-375, Ch. 19)

The operation of one reactor coolant pump or one shutdown cooling (low pressure safety injection) pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control. If no coolant loops are in operation during shutdown operations, suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1.1 or 3.1.1.2, as applicable, is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

←(DRN 03-375, Ch. 19)

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to 272°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against ovepressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

72 hours begins with the discovery of the TEDG unavailability, not to exceed a total of 10 days from the time the EDG originally became inoperable. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

→(DRN 03-375, Ch. 19)

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that(1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. With the minimum AC and DC power sources and associated distribution systems inoperable the ACTION requires the immediate suspension of various activities including operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SHUTDOWN MARGIN or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling concentration. This may result in an overall reduction in boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including increases when operating with a positive moderator temperature coefficient, must also be evaluated to ensure they do not result in a loss of required SHUTDOWN MARGIN. Suspension of these activities does not preclude completion of actions to establish a safe conservative condition.

←(DRN 03-375, Ch. 19)

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that:(1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The K_{eff} value specified in the COLR includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value specified in the COLR also includes a conservative uncertainty allowance of 50 ppm boron.

→(DRN 03-375, Ch. 19)

If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action. Suspension of CORE ALTERATIONS or positive reactivity additions shall not preclude moving a component to a safe position.

←(DRN 03-375, Ch. 19)

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

The equipment door, personnel airlock doors, or penetrations may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided the equipment door, a minimum of one door in the airlock, and penetrations are capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, the equipment door, a minimum of one personnel airlock and the open penetrations will be closed as part of an evacuation of containment. For closure, the equipment door will be held in place by a minimum of four symmetrically-placed bolts.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of CEAs and fuel assemblies, (2) each hoist has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA, and associated handling tool over other irradiated fuel assemblies in the Fuel Handling Building ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

→(DRN 03-375, Ch. 19)

The requirement that at least one shutdown cooling train be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operations.

←(DRN 03-375, Ch. 19)

REFUELING OPERATIONS

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION (Continued)

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the top of the fuel seated in the reactor pressure vessel ensures that a single failure of the operating shutdown cooling train will not result in a complete loss of decay heat removal capability. When there is no irradiated fuel in the reactor pressure vessel, this is not a consideration and only one shutdown cooling train is required to be OPERABLE. With the reactor vessel head removed and 23 feet of water above the top of the fuel seated in the reactor pressure vessel, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 20 REPLACEMENT PAGES
(1 page)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 20 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	Insert
B 2-6	B 2-6

DNBR - Low (Continued)

in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit of 1.26. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	≥ 495°F
b.	RCS Cold Leg Temperature-High	< 580°F
c.	Axial Shape Index-Positive	Not more positive than +0.5
d.	Axial Shape Index-Negative	Not more negative than -0.5
e.	Pressurizer Pressure-Low	≥ 1860 psia
f.	Pressurizer Pressure-High	< 2375 psia
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	≤ 7.00
i.	Quality Margin-Low	> 0

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Reactor Coolant Flow - Low

→ (DRN 03-6)

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a steam line break event with a loss-of-offsite power. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a nominal setpoint of 19.00 psid. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

← (DRN 03-6)

REISSUE

TECHNICAL SPECIFICATION BASES
CHANGE NO. 21 and 22 REPLACEMENT PAGES
(4 pages)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised pages are identified by Change Number 21 or Change No. 22 and contain the appropriate DRN number and vertical lines indicating the areas of change.

Remove	Insert
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3
	B 3/4 9-4

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The K_{eff} value specified in the COLR includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value specified in the COLR also includes a conservative uncertainty allowance of 50 ppm boron.

→(DRN 03-375, Ch. 19)

If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action. Suspension of CORE ALTERATIONS or positive reactivity additions shall not preclude moving a component to a safe position.

←(DRN 03-375, Ch. 19)

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

→ (DRN 03-178, Ch. 21)

In MODES 1, 2, 3 and 4 the escape of radioactivity to the environment is minimized by maintaining containment OPERABLE as described in LCO 3.6.1, " Primary Containment" .

In Mode 6 (REFUELING), the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as " containment

← (DRN 03-178, Ch. 21)

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

→ (DRN 03-178, Ch. 21)

closure" rather than " containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

During CORE ALTERATIONS or movement of irradiated fuel within the containment, the escape of radioactivity to the environment is minimized when the LCO requirements are met.

The equipment door, personnel airlock doors, or penetrations may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided the equipment door, a minimum of one door in the airlock, and penetrations are capable of being closed by an isolation valve, blind flange or manual valve, or capable of being closed on a containment purge isolation signal (CPIS) initiated by the required radiation monitors in the event of a fuel handling accident. An OPERABLE containment purge isolation valve consists of a containment purge valve capable of isolating on manual initiation and on a containment purge isolation test signal from each of the required radiation monitoring instrumentation channels. (Note that Technical Specifications 3/4.3.3, Radiation Monitoring, and 3/4.9.9, Containment Purge Isolation System, are also applicable.) Should a fuel handling accident occur inside containment, the equipment door, a minimum of one personnel airlock door and the open penetrations will be closed. For closure, the equipment door will be held in place by a minimum of four symmetrically-placed bolts. The containment purge lines are automatically closed upon a CPIS if the fuel handling accident releases activity above prescribed levels. Closure of at least on of the containment purge isolation valves is sufficient to provide closure of the penetration. Containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve or blind flange.

← (DRN 03-178, Ch. 21)

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of CEAs and fuel assemblies, (2) each hoist has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA, and associated handling tool over other irradiated fuel assemblies in the Fuel Handling Building ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

→(DRN 03-375, Ch. 19)

The requirement that at least one shutdown cooling train be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operations.

←(DRN 03-375, Ch. 19)

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the top of the fuel seated in the reactor pressure vessel ensures that a single failure of the operating shutdown cooling train will not result in a complete loss of decay heat removal capability. When there is no irradiated fuel in the reactor pressure vessel, this is not a consideration and only one shutdown cooling train is required to be OPERABLE. With the reactor vessel head removed and 23 feet of water above the top of the fuel seated in the reactor pressure vessel, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM (Continued)

→ (DRN 03-233, Ch. 22)

The containment purge valve isolation system consists of the containment purge isolation valves (CAP-103, CAP-104, CAP-203 and CAP-204), the containment purge and exhaust isolation radiation monitors (one required per train as specified in TS 3/4.3.3), the containment purge isolation signal logic and manual isolation logic.

The ACTION statement to close each of the containment purge penetrations may be met by closing at least one valve per penetration (reference Technical Specification 3/4.9.4 and its Basis).

← (DRN 03-233, Ch. 22)

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 23 REPLACEMENT PAGES
(5 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 23 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	Insert
B 3/4 0-4a	B 3/4 0-4a
B 3/4 0-5	B 3/4 0-5
B 3/4 0-6	B 3/4 0-6
-----------	B 3/4 0-7
-----------	B 3/4 0-8

→ (DRN 03-524)

Specification 4.0.1 establishes the requirement that Surveillances must be performed during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual Surveillance Requirements. This specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified interval, in accordance with 4.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated Surveillance Requirements have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:

a. The systems or components are known to be inoperable, although still meeting the Surveillance Requirements or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The Surveillance Requirements associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given Surveillance. In this case, the unplanned event may be credited as fulfilling the performance of the Surveillance. This allowance includes those Surveillances whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by LCO Action Statements do not have to be performed on inoperable equipment because the Action Statements define the remedial measures that apply. Surveillances have to be met and performed in accordance with 4.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with 4.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

← (DRN 03-524)

→ (DRN 03-524)
Some examples of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 750 psig. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High pressure safety injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

← (DRN 03-524)

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

This extension allowed by Specification 4.0.2 is also applicable to Surveillance Requirements required in Technical specification Actions. However, the extension does not apply to the initial performance. The extension only applies to each performance after the initial performance. The initial performance required by the Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single completion time. One reason for not allowing the extension to this completion time is that such an action usually verifies that no loss of function has occurred or accomplishes the function of the inoperable equipment in an alternative manner.

→ (DRN 03-524)

Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within its specified interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified interval was not met.

← (DRN 03-524)

→ (DRN 03-524)

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with required actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with an interval based not on time intervals, but upon specified unit conditions, operational situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period of up to the specified interval to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. Specification 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by required actions.

Failure to comply with specified intervals for surveillance requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified interval is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

← (DRN 03-524)

→ (DRN 03-524)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the allowed outage times of the required actions for the applicable LCO begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the allowed outage times of the required actions for the applicable LCO begin immediately upon the failure of the Surveillance.

Satisfactory completion of the Surveillance within the delay period allowed by this Specification, or within the allowed outage time of the actions, restores compliance with Specification 4.0.1.

← (DRN 03-524)

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillance must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Condition for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 24 REPLACEMENT PAGE(S)
(4 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 24 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	Insert
B 3/4 2-2	B 3/4 2-2
B 3/4 3-3	B 3/4 3-3
B 3/4 7-4a	B 3/4 7-4a
B 3/4 7-5	B 3/4 7-5

→ (DRN 03-656, Ch. 24)

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - Fxy

← (DRN 03-656, Ch. 24)

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^{c}_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^{m}_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic Surveillance Requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provide assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. The LCO requires the maximum azimuthal tilt during normal steady state power operation to be less than or equal to that specified in the COLR. With AZIMUTHAL POWER TILT greater than the limit specified in the COLR, operation is restricted to only those conditions required to identify the cause of the tilt. However, Action item b.2 allows 24 hours to restore the tilt to less than or equal to the limit specified in the COLR following a CEA misalignment event (i.e., CEA drop). A CEA misalignment event causes an asymmetric core power generation and an increase in xenon concentration in the vicinity of the dropped rod. This event may cause the azimuthal tilt to exceed the limit specified in the COLR. The 2 hour action time to reduce core power is not sufficient to recover from the xenon transient. The 24 hour period allows for correction of the misaligned CEA and allows time for the xenon redistribution effects to dampen out due to radioactive decay and absorption. The reduction in xenon concentration (which is aided by operation at full power) will in turn reduce the tilt below the COLR limit.

The 24 hour period is applicable only to a CEA misalignment where the cause of the tilt has been identified. It is based on the time required or the expected xenon transient to dampen out. All other conditions (not due to a CEA misalignment) where the azimuthal tilt exceeds the limit specified in the COLR require action within the specified 2 hours.

The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The Surveillance Requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries in the COLSS and CPCs because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by plant specific documents addressing the recommendations of Regulatory Guide 1.97, as required by Supplement 1 to NUREG-0737, "TMI Action Items." Table 3.3.10 includes most of the plant's RG 1.97 Type A and Category 1 variables. The remaining Type A/Category 1 variables are included in their respective specifications. Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). Category 1 variables are the key variables deemed risk significant because they are needed to: (1) Determine whether other systems important to safety are performing their intended functions; (2) Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and (3) Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

→ (DRN 03-656, Ch. 24)

With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, the inoperable channel should be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring accident monitoring instrumentation during this interval. If the 30 day AOT is not met, a Special Report approved by OSRC is required to be submitted to the NRC within the following 14 days. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Actions. This Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Actions are identified before a loss of functional capability condition occurs.

← (DRN 03-656, Ch. 24)

With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; at least one of the inoperable channels should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information.

Continuous operation with less than the Minimum Channels OPERABLE requirements is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the accident

3/4.7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 27.0 ft Mean Sea Level is based on the maximum elevation at which the levee provides protection, the nuclear plant island structure provides protection to safety-related equipment up to elevation +30 ft Mean Sea Level.

→ (DRN 03-656, Ch. 24)
3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM
← (DRN 03-656, Ch. 24)

3/4.7.6.1 and 3/4.7.6.2 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM

During an emergency, both S-8 units are started to provide filtration and adsorption of outside air and control room envelope recirculated air (reference: FSAR 6.4.3.3). Dosages received after a full power design basis LOCA were calculated to be orders of magnitude higher than other accidents involving radiation releases to the environment (reference: FSAR Tables 15.6-18, 15.7-2, 15.7-4, 15.7-5, 15.7-7).

Acceptable removal efficiency is shown by a methyl iodide penetration of less than 0.5% 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 70%. 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.

The OPERABILITY of this system and control room design provisions are based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

The ACTION to suspend all operations involving movement of irradiated fuel assemblies shall not preclude completion of movement to a safe conservative position.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analysis and is consistent with Regulatory Guide 1.52 and ASTM D3803-1989.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

→ (DRN 03-656, Ch. 24)

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the On-Site Safety Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

← (DRN 03-656, Ch. 24)

Guidance for visual inspection is provided in NRC Generic Letter 90-09. A visual inspection is the observation of the condition of installed snubbers to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or environmental exposure. The functional testing program provides a 95 percent confidence level that 90 to 100 percent of the snubbers will operate within the specified acceptance limits. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber operability.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 25 REPLACEMENT PAGE(S)
(1 page)

6/4/03

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 25 and contains the appropriate DRN number and a vertical line indicating the area of change.

Remove	Insert
B 3/4 6-5	B 3/4 6-5

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM (Continued)

selecting the 18 month frequency were the known reliability of the Cooling Water System, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillances. The flow measurement for the 18 month test shall be done in a configuration equivalent to the accident lineup to ensure that in an accident situation adequate flow will be provided to the containment fan coolers for them to perform their safety function

Verifying that each valve actuates to the full open position provides further assurance that the valves will travel to their full open position on a Safety Injection Actuation Signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

→(DRN 03-666, Ch. 25)
The asterisk "*" footnote associated with the LCO statement allows the opening of closed containment isolation valves on an intermittent basis under administrative controls. The valves within the scope of this footnote include locked or sealed closed containment isolation valves and deactivated automatic containment isolation valves secured in the isolation position. Acceptable administrative controls must include the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.
←(DRN 03-666, Ch. 25)

"Containment Isolation Valves", previously Table 3.6-2, have been incorporated into the Technical Requirements Manual (TRM).

For penetrations with multiple flow paths, only the affected flow path(s) is required to be isolated when a containment isolation valve in that flow path is inoperable. The flow path may be isolated with the inoperable valve in accordance with the Action requirements, provided the leakage rate acceptance criteria, as applicable, is met and controls are in place to ensure the valve is closed. Also, the penetration is required to meed the requirements of GDC-54, and GDC-55 through GDC 57, as applicable, for all the unisolated flow paths.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 26 REPLACEMENT PAGE(S)
(1 page)

6/4/03

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 26 and contains the appropriate DRN number and a vertical line indicating the area of change.

Remove	Insert
B 3/4 5-1d	B 3/4 5-1d

EMERGENCY CORE COOLING SYSTEMS

BASES

ECCS SUBSYSTEMS (Continued)

When in mode 3 and with RCS temperature greater than or equal to 500°F two OPERABLE ECCS subsystems are required to ensure sufficient emergency core cooling capability is available to prevent the core from becoming critical during an uncontrolled cooldown (i.e., a steam line break) from greater than 500°F.

With the RCS temperature below 500°F and the RCS pressure below 1750 psia, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0. The requirement to dissolve a representative sample of TSP in a sample of water borated to be representative of post-LOCA sump conditions provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures. A boron concentration of 3011 ppm boron is postulated to be representative of the highest post-LOCA sump boron concentration. Post LOCA sump pH will remain between 7.0 and 8.1 for the maximum (3011 ppm) and minimum (1504 ppm) boron concentrations calculated using the maximum and minimum post-LOCA sump volumes and conservatively assumed maximum and minimum source boron concentrations.

→ (DRN 02-1635, Ch. 16; DRN 03-445, Ch. 26)

With the exception of systems in operation, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will prevent water hammer, pump cavitation, and pumping noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SIAS or during SDC. The LPSI system has been evaluated for voids in the discharge piping. The piping system has been qualified for the hydraulic transient. In addition, the reactor has been qualified for an intrusion of a small gas bubble. Therefore, from a design basis standpoint, for injection capacity and prevention of water hammer, pump cavitation, and pumping noncondensible gas the LPSI system will be considered operable and full of water with the existence of voids in the system discharge legs. The 31 day frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the adequacy of the procedural controls governing system operation.

← (DRN 02-1635, Ch. 16; DRN 03-445, Ch. 26)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 27 REPLACEMENT PAGE(S)
(2 pages)

6/24/03

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 27 and contain the appropriate DRN number and a vertical line indicating the area of change.

Remove	Insert
B 3/4 3-1c	B 3/4 3-1c
B 3/4 3-2	B 3/4 3-2

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION (Continued)

Response time may be verified by any series of sequential, overlapping, or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the topical report. Respose time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.

TABLE 3.3-1, Functional Unit 13, Reactor Trip Breakers

The Reactor Trip Breakers Functional Unit in Table 3.3-1 refers to the reactor trip breaker channels. There are four reactor trip breaker channels. Two reactor trip breaker channels with a coincident trip logic of one-out-of-two taken twice (reactor trip breaker channels A or B, and C or D) are required to produce a trip. Each reactor trip breaker channel consists of two reactor trip breakers. For a reactor trip breaker channel to be considered OPERABLE, both of the reactor trip breakers of that reactor trip breaker channel must be capable of performing their safety function (disrupting the flow of power in its respective trip leg). The safety function is satisfied when the reactor trip breaker is capable of automatically opening, or otherwise opened or racked-out.

If a racked-in reactor trip breaker is not capable of automatically opening, the ACTION for an inoperable reactor trip breaker channel shall be entered. The ACTION shall not be exited unless the reactor trip breaker capability to automatically open is restored, or the reactor trip breaker is opened or racked-out.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION (Continued)

→(DRN 03-871, Ch. 27)

The Steam Generator Blowdown Process Radiation Monitor and the Component Cooling Water Process Radiation Monitors A, B, and A/B are designed to detect leakage into the monitored system from components that may contain radioactive contamination. These process monitors have an alarm function that annunciates when activity levels at or above the alarm setpoints are detected. This alarm provides an opportunity for the operator to isolate the system and/or equipment and perform investigative activities to locate and repair the source of leakage. By design, the sample flow for these monitors is provided by the hydraulic head established in the monitored system during system operation. When flow in the monitored system is terminated, which would occur if the system was being taken out of service for maintenance, the monitor will go into an alarmed condition due to loss of sample flow. If this alarmed condition is due solely to the termination of the flow in the monitored system, and the process monitors were OPERABLE prior to flow termination, then these radiation monitors should be considered OPERABLE. Therefore, the performance of ACTION 28 is not appropriate or required for this condition. During this condition, the monitors are effectively in a standby state and are capable of automatically performing their intended safety function once flow is re-established in the monitored system. The performance of the shiftly channel check (and other surveillances, if required) should continue during this condition to maintain compliance with the requirements of this Technical Specification.

←(DRN 03-871, Ch. 27)

3/4.3.3.2 INCORE DETECTORS

This section has been deleted.

3/4.3.3.3 SEISMIC INSTRUMENTATION

This section has been deleted.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

This section has been deleted.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 28 REPLACEMENT PAGE(S)
(1 page)

7/21/03

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 28 and contains the appropriate DRN number and a vertical line indicating the area of change.

Remove	Insert
B 3/4 7-8	B 3/4 7-8

3/4.7.12 ESSENTIAL SERVICES CHILLED WATER SYSTEM (Continued)

system to operate in such a manner that ≤ 42°F and/or ≥ 500 gpm may not be directly met, yet CHW System Operability is maintained. During normal operation, when there is insufficient heat load, the following conditions may apply, but the CHW System is still OPERABLE.

(1) The chilled water operational flow control valves for Control Room Ventilation Unit AH-12 and Switchgear Ventilation Units AH-25 and AH-30, control the flow rate through the cooling coils based on discharge air temperature. If there is insufficient load, the flow control valves may be at a minimum, thus, reducing the total chilled water train flow rate to <500 gpm.

2) The CHW System chillers are equipped with a Hot Gas Bypass Valve which opens when chilled water inlet temperature is reduced significantly. This indicates the available heat load on the operating chiller is reduced to a point it will begin to auto recycle if the valve is not opened. This valve diverts a portion of hot compressor discharge gas directly to the bottom of the evaporator instead of sending it to the condenser. This diversion artificially increases the evaporators refrigerant pressure and temperature which in turns increases the chilled water outlet temperature. The increased chilled water outlet temperature eventually increases the chilled water inlet temperature which then closes the Hot Gas Bypass Valve. This operation allows the chiller to stay running at minimum heat loads, down to approximately 10% rated capacity, but allows the chilled water outlet temperature to cycle. Due to this cycling, the peak chilled water outlet temperature may be >42°F. During DBA conditions, air handling unit cooling coil heat loads would be increased which results in the Hot Gas Bypass Valve going to the closed position.

→(DRN 03-1046, Ch. 28)

3) If the Hot Gas Bypass Valve does not open (i.e., is not operational), as described in Item 2, the chiller will auto recycle based on low chilled water outlet temperature. The chiller will automatically secure at a preset low temperature, then automatically restart when the chilled water temperature increases past the reset deadband of the switch. The reset deadband for the switch allows the chilled water outlet temperature to be >42°F. As chiller loading is increased (as would occur during a DBA) the chiller will load sufficiently to reduce chilled water outlet temperature ≤42°F.

←(DRN 03-1046, Ch. 28)

The 31 day Surveillance Requirement (SR) to verify the chilled water outlet temperature is ≤42°F at a flow rate of ≥500 gpm ensures the assumptions of the DBA are preserved. This SR will be performed with sufficient heat load to ensure the Hot Gas Bypass Valve is closed and the chiller is not auto recycling on low load. This may require shifting loads from one chilled train to one being tested. This requirement is reflective of an actual post DBA condition, and ensures the chiller will control the chilled water outlet temperature within limits when sufficient heat load is applied.