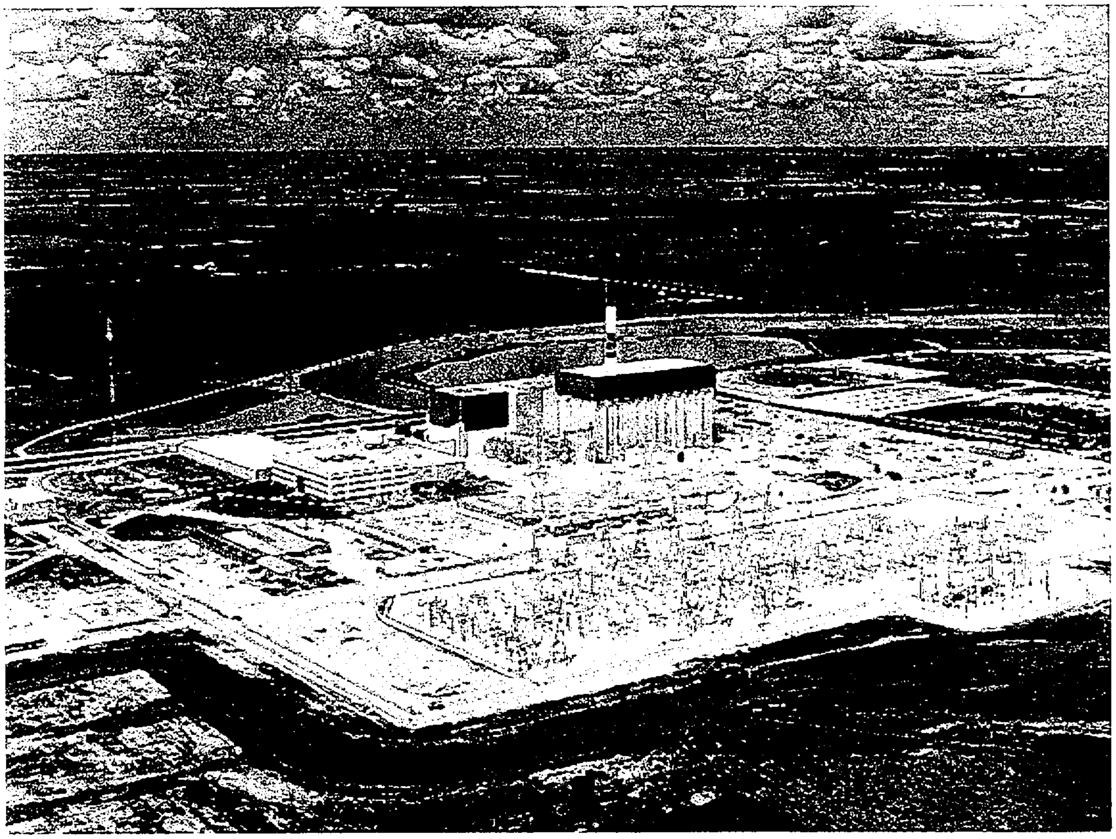
Improved Technical Specifications

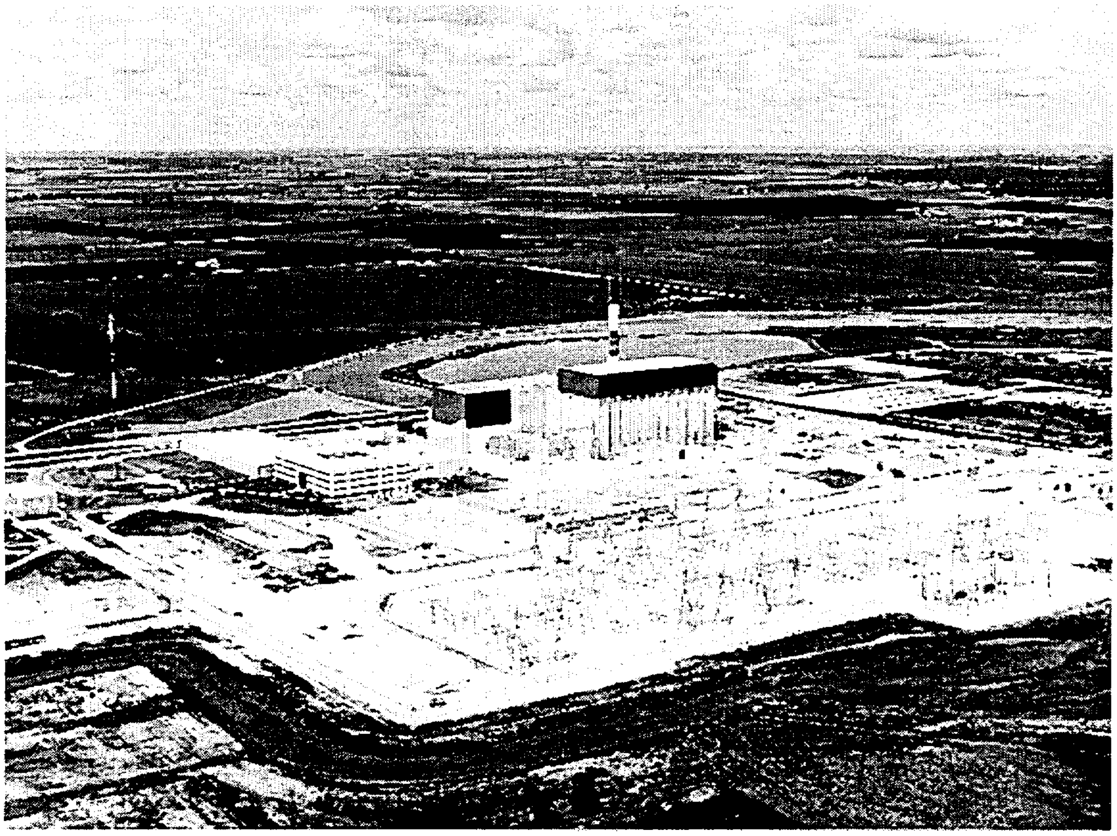


LaSalle County Station

Volume 13:
Unit 2 CTS Markup
In CTS Order

ComEd

Improved Technical Specifications



LaSalle County Station

Volume 13:
Unit 2 CTS Markup
In CTS Order

1.1 A.1 ITS Chapter 1.0

1.0 DEFINITIONS

Note to Definitions

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

of this section and Bases

A.1

ACTIONS

1.1 ACTION S shall be that part of a Specification which that prescribes remedial measures required under designated conditions.

Actions to be taken

within specified Completion Times

A.1

1.2 DELETED

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

1.3 The ~~AVERAGE PLANAR LINEAR HEAT GENERATION RATE~~ (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

LHGRs

at the height

A.1

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

A.1

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment, of channel behavior during operation, by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

to

A.1

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

or actual

required

, interlock, display,

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

A.1

A.3

L.1

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

so

means of

A.1

Page 16 of 29

CORE ALTERATION

A.1

~~1.7~~ CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

A.1

~~1.8~~ The ~~CORE OPERATING LIMITS REPORT~~ COLR is the unit~~-~~specific document that provides ~~core operating~~ cycle specific parameter limits for the current ~~operating~~ reload cycle. These cycle~~-~~specific ~~core operating~~ limits shall be determined for each reload cycle in accordance with Specification ~~6.6.A.6~~ 5.6.5. Plant operation within these ~~operating~~ limits is addressed in ~~individual~~ ~~S~~specifications.

~~CRITICAL POWER RATIO~~

~~1.9~~ The ~~CRITICAL POWER RATIO (CPR)~~ is ~~shall be~~ ~~the ratio of~~ that power in the assembly ~~which~~ that is calculated by application of the appropriate ~~approved CPR~~ correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

Insert into MCPR definition on Page 1-4

DOSE EQUIVALENT I-131

A.1

~~1.10~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram)~~,~~ ~~which~~ that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

L.2

~~E-AVERAGE DISINTEGRATION ENERGY~~

A.2

~~1.11 Ē shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.~~

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

A.1

~~1.12~~ The ~~EMERGENCY CORE COOLING SYSTEM (~~ECCS~~)~~ RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS ~~actuation~~ initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps ~~such~~ so that the entire response time is measured.

A.1

ITS Chapter 1.0

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME (EOC-RPT)

EOC-RPT

A.1

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to energization of the recirculation pump circuit.

Insert 1

A.6

A.1

ITS Chapter 1.0

END-OF-CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) ~~SYSTEM RESPONSE TIME~~ (Continued)

~~breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:~~

~~a. Turbine stop valves, and~~
~~b. Turbine control valves.~~

A.1

The response time may be measured by means of any series of sequential, overlapping, or total steps ~~such~~ so that the entire response time is measured.

~~1.14 DELETED~~ A.1

~~FRACTION OF RATED THERMAL POWER~~

~~1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.~~ A.2

~~FREQUENCY NOTATION~~

~~1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.~~ A.7

~~GASEOUS RADWASTE TREATMENT SYSTEM~~

~~1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~ A.2

~~IDENTIFIED~~ LEAKAGE A.8

A.1 ~~1.18~~ ~~IDENTIFIED~~ LEAKAGE shall be: A.9

1. ~~a.~~ Leakage into the drywell ~~collection systems~~, such as that from pump seals or valve packing ~~leaks~~, that is captured and conducted to a sump or collecting tank; or A.1

A.1

2. ~~b.~~ Leakage into the drywell ~~containment~~ atmosphere from sources that are both specifically located and known either not to interfere with the operation of ~~the~~ leakage detection systems or not to be ~~PRESSURE BOUNDARY~~ LEAKAGE. A.9 A.1 A.8

a. Identified LEAKAGE

ISOLATION SYSTEM RESPONSE TIME

~~1.19~~ The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation ~~actuation~~ setpoint at the channel sensor until the isolation valves travel to their required positions. ~~Times shall include diesel generator starting and sequence loading delays where applicable.~~ The response time may be measured by means of any series of sequential, overlapping, or total steps ~~such~~ so that the entire response time is measured. A.1 A.10 A.1

~~1.20 DELETED~~

INSERT definition of Unidentified LEAKAGE and Total LEAKAGE from page 1-7
INSERT definition of Pressure Boundary LEAKAGE from page 1-5 A.8

A.1

ITS Chapter 1.0

DEFINITIONS

LIMITING CONTROL ROD PATTERN

1.21 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

A.2

LINEAR HEAT GENERATION RATE (LHGR)

~~1.22~~ The ~~LINEAR HEAT GENERATION RATE~~ (LHGR) shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. ~~LHGR is monitored by the ratio of LHGR to its fuel specific limit, as specified in the CORE OPERATING LIMITS REPORT.~~

A.1

LOGIC SYSTEM FUNCTIONAL TEST

~~1.23~~ A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components, (i.e, all relays and contacts, ~~all~~ trip units, solid state logic elements, etc.) of a logic circuit, from sensor ~~through and~~ up to, but not including the actuated device(s) to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps ~~such~~ so that the entire logic system is tested as close to the as practicable.

A.3

A.11

A.1

~~1.24 Deleted~~

L.1

MEMBERS(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

A.2

MINIMUM CRITICAL POWER RATIO (MCPR)

~~1.26~~ The ~~MINIMUM CRITICAL POWER RATIO~~ (MCPR) shall be the smallest critical power ratio (CPR) ~~which~~ that exists in the core for each class of fuel.

A.1

Insert definition of CPR from page 1-2

A.5

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

A.12

moved to Section 5.5

A.1

ITS 5.5

LIMITING CONTROL ROD PATTERN

1.21 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. LHGR is monitored by the ratio of LHGR to its fuel specific limit, as specified in the CORE OPERATING LIMITS REPORT.

See ITS Chapter 1.0

LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e, all relays and contacts, all trip units, solid state logic elements, etc: of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

1.24 Deleted

MEMBERS(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

5.5.1 OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. 5.5.1.a

The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4. 5.5.1.b

A.1

ITS Chapter 1.0

DEFINITIONS

OPERABLE - OPERABILITY

A.1

1.28 A system, subsystem, ~~train~~ division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s), and when all necessary attendant instrumentation, controls, a normal ~~and~~ or an emergency electrical power ~~source~~, cooling ~~or~~ seal water, lubrication, ~~or~~ and other auxiliary equipment that are required for the system, subsystem, ~~train~~, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

A.1 A.13 A.1

~~OPERATIONAL CONDITION - CONDITION~~ MODE

A.13

1.29 A ~~n OPERATIONAL CONDITION, i.e., CONDITION,~~ MODE shall ~~be~~ correspond to any one inclusive combination of mode switch position ~~and~~, average reactor coolant temperature, and reactor vessel head closure bolt tensioning ~~as~~ specified in Table ~~1.2~~ 1.1-1 with fuel in the reactor vessel.

A.14

~~PHYSICS TESTS~~

~~1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.~~

A.2

(d) ~~PRESSURE BOUNDARY~~ LEAKAGE

1.31 ~~PRESSURE BOUNDARY~~ LEAKAGE ~~shall be leakage~~ through a non-isolable fault in a reactor coolant system (RCS) component body, pipe wall, or vessel wall.

A.8

~~PRIMARY CONTAINMENT INTEGRITY~~

~~1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:~~

A.15

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

~~1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or~~

~~2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.6.3.~~

~~b. All primary containment equipment hatches are closed and sealed.~~

~~c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.~~

LA.2 A.15

~~d. The primary containment leakage rates are maintained within the limits per Surveillance Requirement 4.6.1.1.b.~~

~~e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.~~

A.15

A.1

ITS Chapter 1.0

~~PRIMARY CONTAINMENT INTEGRITY (Continued)~~ A.15

~~f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.~~ A.15 LA.2

~~g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.~~ A.15

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

A.16 moved to ITS Chapter 5.0

~~PURGE - PURGING~~

~~1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~ A.2

RATED THERMAL POWER (RTP)

1.35 ~~RATED THERMAL POWER~~ RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT. A.1

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

1.36 ~~REACTOR PROTECTION SYSTEM~~ RPS RESPONSE TIME shall be ~~the~~ that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps ~~such~~ so that the entire response time is measured. A.1

~~REPORTABLE EVENT~~

~~1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.~~ A.2

~~ROD DENSITY~~

~~1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.~~ A.2

CTS 6.7

<See ITS Chapter 1.0>

PRIMARY CONTAINMENT INTEGRITY (Continued)

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

~~PROCESS CONTROL PROGRAM~~

~~1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

LA.1

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

<See ITS Chapter 1.0>

Page 3 of 4

A.1

ITS Chapter 1.0

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

A.15

b. All secondary containment hatches and blowout panels are closed and sealed.

A.15 LA.2

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

A.15

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

A.15 LA.2

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

A.15

SHUTDOWN MARGIN (SDM)

1.40 ~~SHUTDOWN MARGIN~~ SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: c. all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn ~~and the reactor is in the shutdown condition; cold, i.e~~ b. The moderator temperature is 68°F; and a. The reactor is xenon free.

Insert 2

A.1

A.17

SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

A.2

SOURCE CHECK

1.42 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

A.2

STAGGERED TEST BASIS

1.43 A STAGGERED TEST BASIS shall consist of:

A.1

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

A.18

Insert 3

A.1 ITS Chapter 1.0

~~STAGGERED TEST BASIS (Continued)~~

~~b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.~~ A.18

THERMAL POWER

1.44 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. A.1

TURBINE BYPASS SYSTEM RESPONSE TIME

1.45 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps such So that the entire response time is measured. A.1

b. UNIDENTIFIED LEAKAGE

1.46 UNIDENTIFIED All LEAKAGE ~~shall~~ be all leakage which into the drywell that is not IDENTIFIED LEAKAGE; A.8

~~VENTILATION EXHAUST TREATMENT SYSTEM~~

~~1.47 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~ A.2

~~VENTING~~

~~1.48 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~ A.2

A8

c. TOTAL LEAKAGE
SUM OF THE IDENTIFIED AND UNIDENTIFIED LEAKAGE; AND

ITS Chapter 1.0

A.1

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
N.A.	Not applicable.

A.7

ITS Chapter 1.0

A.1

TABLE ~~1.2~~ 1.1-1

~~OPERATIONAL CONDITIONS~~ MODES A.1

~~CONDITION~~ MODE TITLE A.1	REACTOR MODE SWITCH POSITION A.1	AVERAGE REACTOR COOLANT TEMPERATURE °F A.1
1. POWER OPERATION	Run	~~Any temperature~~ NA A.1
2. STARTUP M.1	Refuel(a) or Startup/Hot Standby M.1	~~Any temperature~~ NA
3. HOT SHUTDOWN(a)	Shutdown~~# ***~~ A.20	> 200~~°F~~ A.1
4. COLD SHUTDOWN(a)	Shutdown~~# ## ***~~	≤ 200~~°F~~
5. REFUELING~~*~~(b) A.19	Shutdown or Refuel~~**~~ ~~#~~ A.21	~~≤ 140°F~~ NA M.1

(a) All reactor vessel head closure bolts fully tensioned. M.1

~~#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.~~ A.20 moved to ITS 3.10.1

~~##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.~~ A.20 moved to ITS 3.10.3

(b) ~~*Fuel in the reactor~~ vessel ~~with the~~ One or more reactor vessel head closure bolts less than fully tensioned ~~or with the head removed~~. A.1

~~**See Special Test Exception 3.10.3~~ A.19

~~***The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided that the one-rod-out interlock is OPERABLE.~~ A.21 A.20 moved to ITS 3.10.2 and ITS 3.10.3

add proposed Sections 1.2, 1.3, ~~and~~ 1.4 A.22

A1

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown# ***	> 200°F
4. COLD SHUTDOWN	Shutdown# ## ***	≤ 200°F
5. REFUELING*	Shutdown or Refuel** #	≤ 140°F

<See ITS Chapter 1.0>

Applicability

Applicability of MODE 3, 4, and 5

LCO 3.10.1

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

in core cells containing one or more fuel assemblies. — L1

add proposed LCO 3.10.1.b — L1

or Refuel — L.1

LA1

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exception 3.10.3

***The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided that the one-rod-out interlock is OPERABLE.

<See ITS Chapter 1.0>

Add proposed ACTION and Surveillance Requirements — M.1

A.1

TABLE 1.2

OPERATIONAL CONDITIONS

(See ITS Chapter 1.0)

Applicability

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown# ****	> 200°F
4. COLD SHUTDOWN	Shutdown# ## ***	≤ 200°F
5. REFUELING*	Shutdown or Refuel** #	≤ 140°F

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

***See Special Test Exception 3.10.3

Applicability of MODE 3

LCO 3.10.2 ****The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided that the one-rod-out interlock is

LCO 3.10.2.a OPERABLE.

add proposed LCO 3.10.2.b, c, and d — M.1

add proposed ACTION and Surveillance Requirements — M.1

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown# ***	> 200°F
4. COLD SHUTDOWN	Shutdown# ## ***	≤ 200°F
5. REFUELING*	Shutdown or Refuel** #	≤ 140°F

<See ITS Chapter 1.0>

Applicability

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

Applicability of MODE 4

LCO 3.10.3

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exception 3.10.3

<See ITS Chapter 1.0>

LCO 3.10.3

LCO 3.10.3 b.1

***The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided that the one-rod-out interlock is OPERABLE.

Applicability of MODE 4

add proposed LCO 3.10.3. b.1 control rod position indication requirement M.2

add proposed LCO 3.10.3. b.2 L.2

Page 6 of 6

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

A.1 ITS Chapter 2.0

A.2

2.1 SAFETY LIMITS

moved to ITS 3.3.1.1

THERMAL POWER, Low Pressure or Low Flow

2.1.1 2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2. M.1

ACTION:

2.2 With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4. A.3

THERMAL POWER, High Pressure and High Flow

<LAR300 Pending> A.4

1.12 1.11

2.1.2 2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with two recirculation loop operation and shall not be less than 1.09 with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow. M.2

≥

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2. M.1

ACTION:

<LAR300 Pending> A.4

1.11 ≥ M.2

2.2 With MCPR less than 1.08 with two recirculation loop operation or less than 1.12 1.09 with single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4. A.3

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4. M.1

ACTION:

2.2 With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.4. A.3

ITS Chapter 2.0

A.1

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS A.2

Moved to ITS 3.3.1.1

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.1.3 2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5 M.1

ACTION: L.1

2.2 With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level within 2 hours after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.4. A3

Page 7 of 10

ITS Chapter 2.0

A.1

A.2 moved to ITS 3.3.1.1

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

A.2 moved to ITS 3.3.1.1

ITS 3.3.1.1

A.1

2.2 LIMITING SAFETY SYSTEM SETTINGS — A.11

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

LCO 3.3.1.1 2.2.1 The reactor protection system instrumentation setpoints shall be set consistant with the ~~Trip Setpoint~~ Allowable — LA.1 values shown in Table 2.2.1-1. — A.11

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

ACTIONS A, B, and C With a reactor protection system instrumentation setpoint less A.11 conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status ~~with its setpoint adjusted consistent with the Trip Setpoint value.~~ — LA.7

TABLE .1-1
REACTOR PROTECTION SYSTEM ,NSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120 divisions of full scale	≤ 122 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - Upscale		
1) Two Recirculation Loop Operation		
a) Flow Biased	≤ 0.58W + 59% with a a maximum of	≤ 0.58W + 62% with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	≤ 0.58W + 54.3% with a maximum of	≤ 0.58W + 57.3% with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1043 psig	≤ 1063 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED		
7. Primary Containment Pressure - High	≤ 1.69 psig	≤ 1.89 psig
8. Scram Discharge Volume Water Level - High	≤ 767' 5¼"	≤ 767' 5¼"
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed

---

*See Bases Figure B 3/4 3-1.

moved to ITS 3.3.1.1

A.2

A.1

ITS Chapter 2.0

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Table 3.3.1.1-1

TABLE ~~2.2.1-1~~ A.11

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

	Function ~~FUNCTIONAL UNIT~~	~~TRIP SETPOINT~~ LA.7	ALLOWABLE VALUES LF.1
1.a and 1.b	1. Intermediate Range Monitor, Neutron Flux-High	~~≤ 120 divisions of full scale~~	≤ 122 divisions of full scale
2.a	2. Average Power Range Monitor: a. Neutron Flux-High, Setdown	~~≤ 15% of RATED THERMAL POWER~~	≤ 20% of RATED THERMAL POWER
2.b	b. Flow Biased Simulated Thermal Power - Upscale 1) Two Recirculation Loop Operation a) Flow Biased	~~≤ 0.58W + 59% with a maximum of~~	≤ 0.58W + 62% with a maximum of
	b) High Flow Clamped	~~≤ 113.5% of RATED THERMAL POWER~~	≤ 115.5% of RATED THERMAL POWER
Note (b) to Table 3.3.1.1-1	2) Single Recirculation Loop Operation a) Flow Biased	~~≤ 0.58W + 54.3% with a maximum of~~	≤ 0.58W + 57.3% with a maximum of
	b) High Flow Clamped	~~≤ 113.5% of RATED THERMAL POWER~~	≤ 115.5% of RATED THERMAL POWER
2.c	c. Fixed Neutron Flux-High	~~≤ 118% of RATED THERMAL POWER~~	≤ 120% of RATED THERMAL POWER
3.	3. Reactor Vessel Steam Dome Pressure - High	~~≤ 1043 psig~~	≤ 1063 psig
4.	4. Reactor Vessel Water Level - Low, Level 3	~~≥ 12.5 inches above instrument zero*~~	≥ 11 inches above instrument zero* LA.6
5.	5. Main Steam Line Isolation Valve - Closure	~~≤ 8% closed~~	≤ 12% closed
	6. DELETED		
6.	7. Primary Containment Pressure - High	~~≤ 1.69 psig~~	≤ 1.89 psig
7.a and 7.b	8. Scram Discharge Volume Water Level - High	~~≤ 767' 5¼"~~	≤ 767' 5¼"
8.	9. Turbine Stop Valve - Closure	~~≤ 5% closed~~	≤ 7% closed

~~*See Bases Figure B 3/4 3-1.~~ A.12

ITS Chapter 2.0

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 414 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.
13. Control Rod Drive		
a. Charging Water Header Pressure-Low	≥ 1157 psig	≥ 1134 psig
b. Delay Timer	≤ 10 seconds	≤ 10 seconds

moved to ITS 3.3.1.1 A2

A.1

Table 3.3.1.1-1

TABLE 2.2.1-1 A.11

Function

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

LF.1

FUNCTIONAL UNIT	~~**TRIP SETPOINT**~~ LA.7	**ALLOWABLE VALUES**
9. 10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	~~≥ 500 psig~~	≥ 414 psig
10. 11. Reactor Mode Switch Shutdown Position	~~N.A.~~	N.A.
11. 12. Manual Scram	~~N.A.~~	N.A.
~~13. Control Rod Drive~~		
~~a. Charging Water Header Pressure-Low~~	~~≥ 1157 psig~~	~~≥ 1134 psig~~
~~b. Delay Timer~~	~~≤ 10 seconds~~	~~≤ 10 seconds~~ R.1

ITS 3.3.1.1

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A.1

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION (LCO)

A.2

A.2

(LCO 3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specification 3.0.6.

Insert 1

A.3

A.2

(LCO 3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in Specification 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

Insert 2

A.4

A.2

(LCO 3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This specification is not applicable in OPERATIONAL CONDITION 4 or 5.

Insert 3

A.5

A.2

(LCO 3.0.4 Entry into an OPERATIONAL CONDITION or other specified CONDITION shall not be made when the conditions for the Limiting Conditions for Operations are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL CONDITION may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

Insert 4

A.6

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in OPERATIONAL CONDITION 4 or 5.

A.7

⟨moved to ITS 3.8.1⟩

A.2

LCO 3.0.5

A.1

3.0.6 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to ~~Specification 3.0.1 and~~ 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO

A.3

add proposed LCO 3.10.6 — A.8

add proposed LCO 3.10.7 — A.9

add proposed LCO 3.0.8 — A.14

ITS 3.8.1

3/4.0 APPLICABILITY

A.1

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specification 3.0.6.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in Specification 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This specification is not applicable in OPERATIONAL CONDITION 4 or 5.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified CONDITION shall not be made when the conditions for the Limiting Conditions for Operations are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL CONDITION may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

(See ITS 3.0)

Required Actions A.2, B.2, C.2, and D.1

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours ~~action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:~~

~~1. At least STARTUP within the next 6 hours,~~
~~2. At least HOT SHUTDOWN within the following 6 hours, and~~
~~3. At least COLD SHUTDOWN within the subsequent 24 hours.~~

L.17

24 hours for proposed Required Action A.2
12 hours for proposed Required Action D.1
4 hours for proposed Required Actions B.2 and C.2

~~This specification is not applicable in OPERATIONAL CONDITION 4 or 5.~~ A.16

3.0.6 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to Specification 3.0.1 and 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

declare required features inoperable. L.17 (See ITS 3.0)

ITS Section 3.0

A.1

APPLICABILITY

3.0 SURVEILLANCE REQUIREMENTS (SR)

A.2

SR 3.0.1 4.0.1 ~~Surveillance Requirements~~ SRs shall be met during the ~~OPERATIONAL CONDITIONS~~ MODES or other conditions ~~specified~~ in the Applicability for individual ~~Limiting Conditions for Operation~~ LCOs unless otherwise stated in ~~an individual Surveillance Requirement~~. Insert 5 A.10

A.2

SR 3.0.2 4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. Insert 6 A.11 M.1 L.1

SR 3.0.3 4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. A.10 ~~The time limits of the ACTION requirements are applicable at the time it is~~ identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Insert 7 L.2 SR 3.0.1 ~~Surveillance Requirements do not have to be performed on inoperable equipment.~~ A.10

A.2

SR 3.0.4 4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable CONDITION shall not be made unless the Surveillance Requirements associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Insert 8 A.12

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. 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 Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

A.13

moved to ITS Section 5.5

← add proposed SR 3.0.5 A.14

A.1

ITS 5.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

See ITS Section 3.0

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable CONDITION shall not be made unless the Surveillance Requirements associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements.

5.5.7 4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows: LA.4

pumps and valves LA.4

a. 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 Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). LA.5 LA.4

5.5.7.a b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days
Every 48 months	At least once per 1461 days

A.5

ITS Section 3.0

A.1

APPLICABILITY SR

3.0 SURVEILLANCE REQUIREMENTS (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

A.13

<Moved to ITS Section 5.5>

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A.1

ITS 5.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.b c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities. LA.4

~~d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.6

5.5.7.d e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

~~f. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.~~ LA.4

5.5.7.c The provisions of SR 3.0.3 are applicable to inservice testing activities; and A.2

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A.1

ITS Chapter 1.0

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN, and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

<See ITS 3.1.1>

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

SHUTDOWN MARGIN

A.7

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A.1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

LCO 3.1.1 3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

ACTION A
ACTION B
a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTIONS C and D
ACTION D
b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. A.2 A.3 A.4 A.5

ACTION E
c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. L.1 A.3 A.5 L.2 A.7 A.4 L.2 A.3

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

SR 3.1.1.1
a. By measurement, ~~prior to or during the first startup~~ after each refueling. A.8 A.9

~~b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.~~ L.3

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod. A.10 moved to ITS 3.1.3

← add proposed first Frequency to S.R. 3.1.1.1 M.1

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ITS 3.1.3

3/4.1 REACTIVITY CONTROL SYSTEMS

A.1

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN, and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle: ⟨See ITS 3.1.1⟩

Required Action A.4

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

Required Action A.4

c. Within (12) (72) hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

L.7

A.5

A.10

⟨Moved to ITS Chapter 1.0⟩

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ITS 3.1.2

A.1

REACTIVITY CONTROL SYSTEM

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

LCO 3.1.2 3.1.2 The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall not exceed 1% delta k/k. A.2

Core keff M.1

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% delta k/k: L.1

ACTION A a. Within ~~12~~ 72 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected. LA.1

ACTION B b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

A.2

SR 3.1.2.1 4.1.2 The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall be verified to be less than or equal to 1% delta k/k: M.1 A.3

a. During the first startup following CORE ALTERATIONS, and L.2

b. At least once per 31 effective full power days during POWER OPERATION. L.3

1000 MWD/T

Core keff M.1

page 2 of 2

ITS 3.1.3

A.1

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

<general reorganization> A.2

LCO 3.1.3 3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2. A.3

ACTION: add proposed ACTIONS NOTE A.4

add proposed Required Action A.1 Note

ACTION A a. With one control rod inoperable due to ~~being immovable, as a result of~~ excessive friction or mechanical interference, or known to be untrippable: A.5

1. Within (1) hour:

(ACTION D a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions. M.1 L.1 M.2

Required Actions A.2 and A.4 L.2

b) Disarm the associated ~~directional~~ control valves* either: control rod drive (CRD)

~~1) Electrically, or~~

2) Hydraulically by closing the drive water and exhaust water isolation valves. LA.1

c) Comply with Surveillance Requirement 4.1.1.c. add proposed Required Action A.1

ACTION E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours. L.3

3. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours. M.2

ACTION C b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above: add proposed ACTION B

1. If the inoperable control rod(s) is withdrawn:

a) Immediately verify: L.4

(ACTION D 1) That the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rod(s) by at least two control cells in all directions, and L.1 M.1

2) The insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range**. M.3

Required Actions C.1 and C.2

b) Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated ~~directional control valves* either:~~ CRD

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves. LA.1

add proposed Required Action C.1 Note A.4

~~*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~ A.6

~~**The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.~~ M.3

A.1

ITS 3.1.3

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

Action C 2. If the inoperable control rod(s) is inserted: L.4

a) Within ~~1~~ 4 hour disarm the associated directional control valves either: CRD LA.1

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

Action E b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

Action E c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d*. With one or more SDV vent or drain lines with one valve inoperable, A.7 moved to ITS 3.1.8

1. Isolate** the associated line within 7 days.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

e*. With one or more SDV vent or drain lines with both valves inoperable,

1. Isolate** the associated line within 8 hours.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by: A.7 moved to ITS 3.1.8

a. At least once per 31 days verifying each valve to be open**, and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

SR3.1.3.2
SR3.1.3.3 4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods ~~not~~ required to have ~~their~~ directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch: A.8 M.5

a. At least once per 7 days, and L.5

Required Action A.3 b. ~~At least once per~~ 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference. L.6

---

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.6

**These valves may be closed intermittently for testing under administrative control. A.7 moved to ITS 3.1.8

*Separate Action statement entry is allowed for each SDV vent and drain line.

**An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

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A.1

ITS 3.1.8

add proposed LCO and Applicability A.2

ACTION (Continued)

See ITS 3.1.3

2. If the inoperable control rod(s) is inserted:

   a) Within 1 hour disarm the associated directional control valves[*] either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

   b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

Action A

d[#]. With one or more SDV vent or drain lines with one valve inoperable,

1. Isolate[##] the associated line within 7 days.

Action C

2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

Action B

e[#]. With one or more SDV vent or drain lines with both valves inoperable,

1. Isolate[##] the associated line within 8 hours.

Action C

2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

SR 3.1.8.1 a. At least once per 31 days verifying each valve to be open[**], and

SR 3.1.8.2 b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

See ITS 3.1.3

4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

---

[*] May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

Note to SR 3.1.8.1

[**] These valves may be closed intermittently for testing under administrative control.

Note 1 to Actions

[#] Separate Action statement entry is allowed for each SDV vent and drain line.

Note 2 to Actions

[##] An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

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ITS 3.1.3

A.1

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

~~4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.~~ A.9

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per 18 months by verifying that the drain and vent valves:

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open after the scram signal is reset.

A.7 moved to LCO 3.1.8

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A.1

ITS 3.1.8

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

See ITS 3.1.3

SR 3.1.8.3 4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per ~~18~~ 24 months by verifying that the drain and vent valves:

LD.1

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open after the scram signal is reset.

actual or simulated

A.3

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ITS 3.1.3

REACTIVITY CONTROL SYSTEM

A.1

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

<general organization> A.11

SR 3.1.3.4 3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero shall not exceed 7.0 seconds. A.12

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION A or C With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds. L.8

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

add proposed SR 3.1.3.4 A.13

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS* or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

<See ITS 3.1.4>

---

*Except normal control rod movement

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REACTIVITY CONTROL SYSTEM

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

A.1

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

See ITS 3.1.3

SURVEILLANCE REQUIREMENTS

SR3.1.4.1
SR3.1.4.2
SR 3.1.4.4

800 — M.1

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

Note to Surveillance Requirements

SR3.1.4.4
SR 3.1.4.1

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

prior to exceeding 40% RTP — M.2

SR 3.1.4.4

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

add proposed SR 3.1.4.3 — M.2

SR 3.1.4.2

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

LA.1

SR 3.1.4.4 Except normal control rod movement. — L.1

Page 4 of 6

A.1

ITS 3.1.4

REACTIVITY CONTROL SYSTEM
CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.1.4 and Table 3.1.4-1

M.3

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Footnote(c) to Table 3.1.4-1

<0531>

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

Action A With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4

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ITS 3.1.4

A.1

REACTIVITY CONTROL SYSTEM

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.1.4 and Table 3.1.4-1

M.3

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest ~~control rods in each group of four control rods arranged in a~~ two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, ~~shall not exceed any of the following:~~

Footnote(c) to Table 3.1.4-1

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION A With the average scram insertion times of control rods exceeding the above limits:

M.3

1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.4

Page 6 of 6

A.1

ITS 3.1.5

REACTIVITY CONTROL SYSTEM

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

LCO 3.1.5 3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*. — A.2 moved to ITS 3.9.5 and ITS 3.10.7

ACTION: — add proposed ACTIONS Note — A.3

a. In OPERATIONAL CONDITION 1 or 2:

ACTION A 1. With one control rod scram accumulator inoperable: and reactor pressure ≥ 900 psig — M.1

a) Within 8 hours, either:

1) ~~Restore the inoperable accumulator to OPERABLE status,~~ — A.4 ~~or~~

add proposed Required Action A.1 — L.1

2) Declare the control rod associated with the inoperable accumulator inoperable.

b) ~~Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.~~ — A.5

A.1

ACTIONS B and C 2. With ~~more than one~~ control rod scram accumulator inoperable, declare the associated control rod inoperable and: within 1 hour — L.1

add proposed Required Action B.2.1

Required Action D.1 Note — a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD pump is operating ~~by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range~~ or place the reactor mode switch in the Shutdown position. — L.2 — A.6

ACTION D

~~b) Insert the inoperable control rods and disarm the associated directional control valves either:~~ — A.7

~~1) Electrically, or~~

~~2) Hydraulically by closing the drive water and exhaust water isolation valves.~~

~~Otherwise, be in at least HOT SHUTDOWN within 12 hours.~~

b. In OPERATIONAL CONDITION 5 with: — A.2 moved to ITS 3.9.5 and ITS 3.10.7

1. One withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within 1 hour, either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

2. More than one withdrawn control rod with the associated scram accumulator inoperable or with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

---

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Page 3 of 4

CONTROL ROD SCRAM ACCUMULATORS

A.1

ITS 3.9.5

LIMITING CONDITION FOR OPERATION

LCO 3.9.5 3.1.3.5 All control rod scram accumulators shall be OPERABLE. A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*. Add proposed control rod scram insertion capability M.1

ACTION:

See ITS 3.1.5

a. In OPERATIONAL CONDITION 1 or 2:

1. With one control rod scram accumulator inoperable:

   a) Within 8 hours, either:

      1) Restore the inoperable accumulator to OPERABLE status, or

      2) Declare the control rod associated with the inoperable accumulator inoperable.

   b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. With more than one control rod scram accumulator inoperable, declare the associated control rod inoperable and:

   a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD pump is operating by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range or place the reactor mode switch in the Shutdown position.

   b) Insert the inoperable control rods and disarm the associated directional control valves either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5 with: Add proposed ACTION A for control rod scram insertion capability m.1

ACTION A 1. One withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod ~~and disarm the associated directional control valves within 1 hour, either:~~ A.4

   ~~a) Electrically, or~~

   ~~b) Hydraulically by closing the drive water and exhaust water isolation valves.~~

A.5 Moved to ITS 3.10.7 2. More than one withdrawn control rod with the associated scram accumulator inoperable or with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

A.2

*At least the accumulator associated with each withdrawn control rod. ~~Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ A.3

A.1 ITS 3.10.7

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*. <See ITS 3.9.5>

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one control rod scram accumulator inoperable:

   a) Within 8 hours, either:

      1) Restore the inoperable accumulator to OPERABLE status, or

      2) Declare the control rod associated with the inoperable accumulator inoperable.

   b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours. <See ITS 3.1.5>

2. With more than one control rod scram accumulator inoperable, declare the associated control rod inoperable and:

   a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD pump is operating by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range or place the reactor mode switch in the Shutdown position.

   b) Insert the inoperable control rods and disarm the associated directional control valves either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5 with: M.1

1. One withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within 1 hour, either: <See ITS 3.9.5>

   LCO 3.10.7.f and ACTION 8

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

2. More than one withdrawn control rod with the associated scram accumulator inoperable or with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position. A.7 M.1

   LCO 3.10.7.f and ACTIONS

---

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2. <See ITS 3.9.5>

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A.1

ITS 3.1.5

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

SR3.1.5.1 a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig, unless the control rod is inserted and disarmed or scrammed. A8

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

SR 3.9.5.2 — a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed. A.6

Add proposed SR 3.9.5.1 M.1

ITS 3.1.3

A.1

REACTIVITY CONTROL SYSTEM

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

SR3.1.3.5 3.1.3.6 All control rods shall be coupled to their drive mechanisms. A.14

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, ~~and 5*~~. L.9

ACTION:

Action C a. In OPERATIONAL CONDITIONS 1 and 2 with one control rod not coupled to its associated drive mechanism: L.4

1. Within 2 hours, either: L.10

a) ~~If permitted by the RWM~~, insert the ~~control rod drive mechanism to accomplish recoupling~~ ~~and verify recoupling by withdrawing the control rod, and:~~ A.15 L.11

1) ~~Observing any indicated response of the nuclear instrumentation, and~~ L.12

2) ~~Demonstrating that the control rod will not go to the overtravel position.~~ L.11

Action C b) ~~If recoupling is not accomplished on the first attempt or, if not permitted by the RWM then until permitted by the RWM,~~ declare the control rod inoperable and insert the control rod and disarm the associated ~~directional control valves either:~~ CRD LA.1 L.10

~~1) Electrically, or~~

~~2) Hydraulically by closing the drive water and exhaust water isolation valves.~~

Action E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours. L.9

~~b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours, either:~~

~~1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or~~

~~2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:~~

~~a) Electrically, or~~

~~b) Hydraulically by closing the drive water and exhaust water isolation valves.~~

~~*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ L.9

**~~May be rearmed intermittently, under administrative control~~, to permit testing associated with restoring the control rod to OPERABLE status. A.6

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ITS 3.1.3

A.1

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS

SR3.1.3.5 4.1.3.6 A control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position: L.12

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity, A.16

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

SR 3.1.3.1 3.1.3.7 The control rod position indication system shall be OPERABLE. A.17

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*. A.18 moved to ITS 3.9.4

ACTION:

Action C a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable within one hour:

1. Determine the position of the control rod by:
   - (a) ~~Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,~~ LA.2
   - (b) ~~Returning the control rod, by single notch movement, to its original position, and~~
   - (c) ~~Verifying no control rod drift alarm at least once per 12 hours, or~~ L.13
2. ~~Move the control rod to a position with an OPERABLE position indicator, or~~ LA.2
3. When THERMAL POWER is:
   - (a) Within the low power setpoint of the RWM:
     - (1) Declare the control rod inoperable. M.6
     - (2) Verify the position and bypassing of control rod with inoperable "Full in" and/or "Full out" position indicators by a second licensed operator or other technically qualified member of the unit technical staff. A.19

   Required Action C.1 Note

   - b) Greater than the low power setpoint of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated ~~directional control valves** either:~~ CRD LA.1
     - (1) ~~Electrically, or~~
     - (2) ~~Hydraulically by closing the drive water and exhaust water isolation valves.~~

Action E 4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

---

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2. A.18 moved to ITS 3.9.4

~~**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~ A.6

ITS 3.9.4

REACTIVITY CONTROL SYSTEM

CONTROL ROD POSITION INDICATION

A.1

LIMITING CONDITION FOR OPERATION

"full-in" Channel

LCO 3.9.4 3.1.3.7 The control rod position indication ~~system~~ shall be OPERABLE. L-1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5~~*~~. M.1

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable within one hour:

1. Determine the position of the control rod by:
    (a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,
    (b) Returning the control rod, by single notch movement, to its original position, and
    (c) Verifying no control rod drift alarm at least once per 12 hours, or
2. Move the control rod to a position with an OPERABLE position indicator, or
3. When THERMAL POWER is:
    (a) Within the low power setpoint of the RWM:
        (1) Declare the control rod inoperable,
        (2) Verify the position and bypassing of control rod with inoperable "Full in" and/or "Full out" position indicators by a second licensed operator or other technically qualified member of the unit technical staff.
    b) Greater than the low power setpoint of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:
        (1) Electrically, or
        (2) Hydraulically by closing the drive water and exhaust water isolation valves.
4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

See ITS 3.1.3

~~*At least each withdrawn control rod.~~ ~~Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ M.1 A.2

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

A.18 moved to ITS 3.9.4

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

a. At least once per 24 hours that the position of each control rod is indicated,

~~b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and~~

~~c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.~~

~~d. That the control rod position indicator corresponds to the control rod position indicated by the "Full in" position indicator:~~

~~1. Prior to each reactor startup, and~~

~~2. Each time a control rod is fully inserted.~~

L.14

---

*At least each withdrawn control rod not applicable to control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

A.18 moved to ITS 3.9.4

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A.1

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

M.1 full-in L.1

ACTION A b. In OPERATIONAL CONDITION 5~~*~~ with a ~~withdrawn~~ control rod full-in position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod. M.1

Add proposed ACTIONS NOTE A.3

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 4.1.3.7 The control rod full-in position indication ~~system~~ Channel shall be determined OPERABLE L.1 by verifying:

~~a. At least once per 24 hours that the position of each control rod is indicated,~~

~~b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and~~

~~c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.~~

~~d. That the control rod position indicator corresponds to the control rod position indicated by the "Full in" position indicator:~~ L.1

~~1. Prior to each reactor startup, and~~

~~2. Each time a control rod is fully inserted.~~

M.1 A.2

~~*At least each withdrawn control rod not applicable to control rods removed per Specifications 3.9.10.1 or 3.9.10.2.~~

LTS 3/4.1.3.8

REACTIVITY CONTROL SYSTEM

~~CONTROL ROD DRIVE HOUSING SUPPORT~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.3.8 The control rod drive housing support shall be in place.~~

~~**APPLICABILITY:** OPERATIONAL CONDITIONS 1, 2 and 3.~~

~~**ACTION:**~~

~~With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.1.3.8 The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.~~

L.1

Page 2of2

A.1

ITS 3.3.2.1

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

LCO 3.3.2.1
Table 3.3.2.1-1
Function 2

3.1.4.1 The rod worth minimizer (RWM) shall be OPERABLE. M.3

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*, when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER, the minimum allowable low power setpoint.

Add proposed Required Actions C.2.1.1 and C.2.1.2 — L.2

ACTION:

Conditions C + D

Required Actions C.2.2 and D.1

Required Action C.1

a. With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

SR 3.3.2.1.9

b. With an inoperable control rod(s), OPERABLE control rod movement may continue by bypassing the inoperable control rod(s) in the RWM provided that:

1. The position and bypassing of inoperable control rods is verified by a second licensed operator or other technically qualified member of the unit technical staff, and

~~2. There are not more than 3 inoperable control rods in any RWM group.~~ L.3

c. ~~The provisions of Specification 3.0.4 are not applicable, with the exception that control rod withdrawal for reactor startup shall not begin with the RWM inoperable.~~ A.3 L.2

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE: L.4

SR 3.3.2.1.2 AND NOTE
SR 3.3.2.1.3 AND NOTE

a. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 prior to reaching 10% of RATED THERMAL POWER when reducing THERMAL POWER, ~~by verifying proper annunciation of the selection error of at least one out-of-sequence control rod.~~ LA.2

NOTE TO SR 3.3.2.1.2

*Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality. M.3

REACTIVITY CONTROL SYSTEM

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

A.1

ROD WORTH MINIMIZER

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.3.2.1.2 AND NOTE b. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function ~~by demonstrating inability to withdraw an out-of-sequence control rod.~~ L.4 LA.2

SR 3.3.2.1.3 AND NOTE c. In OPERATIONAL CONDITION 1 within one hour after RWM automatic initiation when reducing THERMAL POWER, ~~by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.~~ L.4 LA.2

SR 3.3.2.1.8 d. By verifying the control rod patterns and sequence input to the RWM computer is correctly loaded following any loading of the program into the computer.

Add proposed SR 3.3.2.1.6 M.4

REACTIVITY CONTROL SYSTEM

A.1

ITS 3.3.2.1

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

LCO 3.3.2.1
Table 3.3.2.1-1
Function 1

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and no peripheral control rod is selected. A.2

ACTION:

L.5

Action A, Action B — a. With one RBM channel inoperable, ~~verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN and~~ restore the inoperable RBM channel to OPERABLE status within 24 hours; otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

Action B — b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

SR 3.3.2.1.1
SR 3.3.2.1.4

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

~~b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.~~

L.5

REACTIVITY CONTROL SYSTEM

A.1

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.1.7 3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, ~~and 5*~~. L.1

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

ACTION A, ACTION C 1. With one motor operated suction valve, one pump and/or one explosive valve inoperable, restore the inoperable suction valve, pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION B, ACTION C 2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

~~b. In OPERATIONAL CONDITION 5*:~~

~~1. With one motor operated suction valve, one pump and/or one explosive valve inoperable, restore the inoperable suction valve, pump and/or explosive valve to OPERABLE status within 30 days or insert all insertable control rods within the next hour.~~

~~2. With the standby liquid control system inoperable, insert all insertable control rods within 1 hour.~~ L.1

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

SR 3.1.7.1, SR 3.1.7.2 1. The available volume and temperature of the sodium pentaborate solution are within the limits of Figures 3.1.5-1 and 3.1.5-2, and

SR 3.1.7.3 2. ~~The heat tracing circuit is OPERABLE by~~ verifying the ~~indicated~~ temperature of the pump suction piping up to the storage tank outlet valves are ~~to be~~ ≥ ~~80~~ 68°F ~~on the local indicator~~. A.2 LA.1 M.1

~~*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ L.1

ITS 3.1.7

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. ~~Starting both pumps and recirculating demineralized water to the test tank.~~ L.2

SR 3.1.7.4 2. Verifying the continuity of the explosive charge.

SR 3.1.7.5 3. Determining that the concentration of boron in solution is within the limits of Figure 3.1.5-2 by chemical analysis.*

SR 3.1.7.6 4. Verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position. or can be aligned to the correct position A.3

c. At least once per ~~18~~ 24 LD.1 months ~~during shutdown~~ L.3 by;

1. Initiating one of the standby liquid control system loops, ~~including an explosive valve~~ LA.2 and verifying that a flow path from the pumps to the reactor pressure vessel is available ~~by pumping demineralized water into the reactor vessel~~ LA.3. ~~The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired.~~ LA.2 Both injection loops shall be tested in ~~36~~ 48 LD.1 months.

SR 3.1.7.8

Definition of STAGGERED TEST BASIS

SR 3.1.7.7 2. Demonstrating that when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1220 psig is met.

3. ~~Demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank.~~ LA.4

A.4

A.5

SR 3.1.7.9 4. ~~**Demonstrating that all heat traced piping between the storage tank and the reactor vessel is~~ storage tank outlet valves are unblocked ~~by verifying flow from the storage tank to the motor operated suction~~ storage tank outlet A.8 valve ~~and then draining and flushing the piping with demineralized water.~~ LA.3

5. ~~Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized.~~ L.4

SR 3.1.7.5 *This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

~~**This test shall also be performed whenever the heat tracing circuit has been found to be inoperable~~ A.5 ~~and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.~~ A.6

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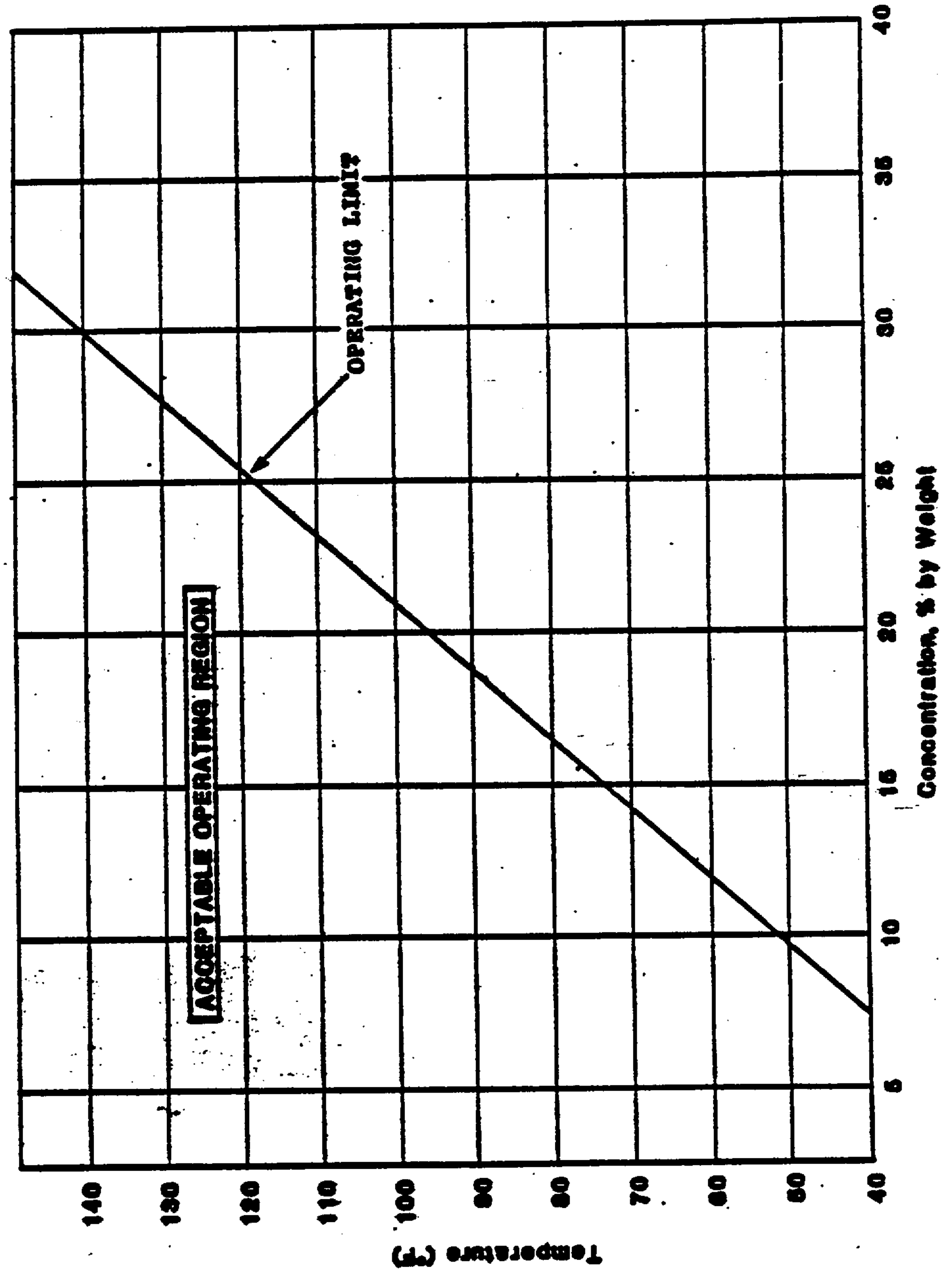
ITS 3.1.7

A.1



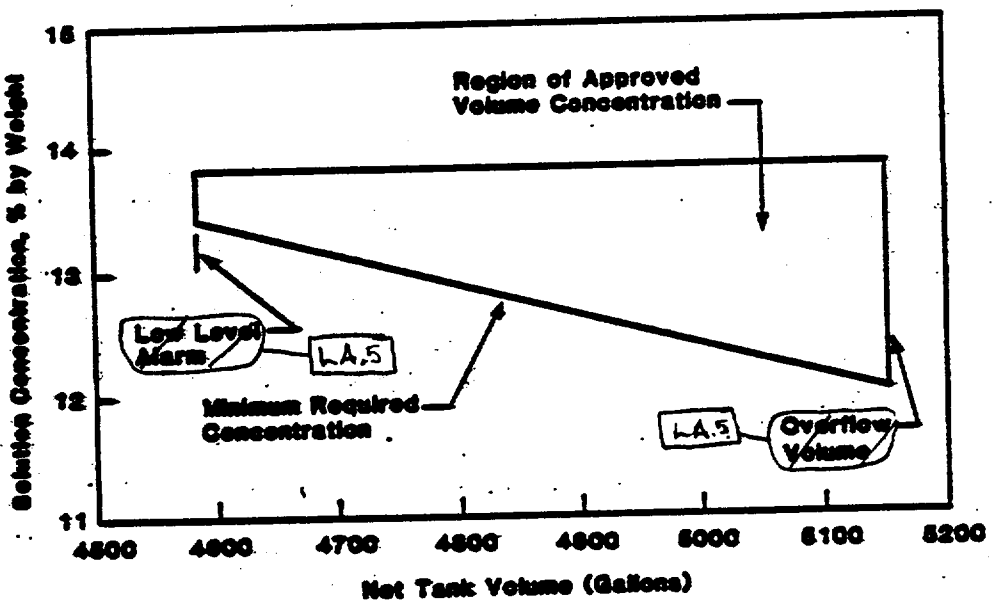
SODIUM PENTABORATE SOLUTION TEMPERATURE / CONCENTRATION REQUIREMENTS

~~Figure 3.1.5-1~~

Figure 3.1.7-2

ITS 3.1.7

A.1



SODIUM PENTABORATE ($Na_2B_{10}O_{16}$ $10H_2O$)
VOLUME/CONCENTRATION REQUIREMENTS

Figure 3.1.5-2

Figure 3.1.7-1

CTS 3/4.1.6

REACTIVITY CONTROL SYSTEM

~~3/4.1.6 ECONOMIC GENERATION CONTROL SYSTEM~~

~~LIMITING CONDITION FOR OPERATION~~

R.1

~~3.1.6 The economic generation control system may be in operation with automatic flow control provided that:~~

~~a. Core flow is ≥ 65% of rated core flow, and~~

~~b. THERMAL POWER is greater than or equal to 20% of RATED THERMAL POWER.~~

~~APPLICABILITY: OPERATIONAL CONDITION 1.~~

~~ACTION: With core flow less than 65% of rated core flow or THERMAL POWER less than 20% of RATED THERMAL POWER, cease operation under the economic generation control system.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.1.6 The economic generation control system shall be demonstrated OPERABLE by:~~

~~a. Calculating current efficiency and, using a nominal curve of effiency versus THERMAL POWER, verifying that the EGC lower MW_e setpoint will maintain core flow > 65% of rated core flow and THERMAL POWER ≥ 20% of RATED THERMAL POWER:~~

~~1. Prior to entry into EGC operation, and~~

~~2. At least once per 12 hours while operating in EGC.~~

~~b. Verifying that current core flow is > 65% of rated core flow and THERMAL POWER is ≥ 20% of RATED THERMAL POWER:~~

~~1. Prior to entry into EGC operation, and~~

~~2. At least once per 12 hours while operating in EGC.~~

Page 2 of 2

ITS 3.2.1

3/4.2 POWER DISTRIBUTION LIMITS A1

3/4.2.1 AVERAGE PLANAR LINEAR HEAT-GENERATION RATE

LIMITING CONDITION FOR OPERATION

LCO 3.2.1 3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: ~~OPERATIONAL CONDITION 1~~ when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. A2

ACTION:

Action A — With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, ~~initiate corrective action within 15 minutes and~~ restore APLHGR to within the required limits within 2 hours LA1

Action B — or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

SR 3.2.1.1

a. At least once per 24 hours. L.1

b. Within 12 hours after ~~completion of a~~ THERMAL POWER ~~increase of at least 15%~~ ≥25% of RATED THERMAL POWER, and L.1

c. ~~Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.~~ L.2

Page 2 of 2

ITS 3.2.2

POWER DISTRIBUTION LIMITS

A.1

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. A.2

ACTION

Action A
a. With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective action within 15 minutes, and

~~2. Restore MCPR to within the required limit within 2 hours.~~ LA.1

Action B
3. Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

Action A
b. When operating in a condition not specified in the CORE OPERATING LIMITS REPORT, restore within 2 hours reduce THERMAL POWER to less than 25% of RATED THERMAL
Action B
POWER within 4 hours. L.1

Specification 3.3.4.1

A.1

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

LCO 3.3.4.1.b

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

~~OPERATIONAL CONDITION 1, when~~ THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. — A.3

ACTION

a. With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective action within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.
3. Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

(SEE ITS 3.2.2)

Required Action C.2 b. When operating in a condition not specified in the CORE OPERATING LIMITS REPORT, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within 4 hours.

Add Required Action C.1 — L.2

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ITS 3.2.2

A.1

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

SURVEILLANCE REQUIREMENTS

SR3.2.2.1 4.2.3.1 MCPR shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours,

b. Within 12 hours after ~~completion of a~~ THERMAL POWER ≥25% ~~increase of at least 15%~~ of RATED THERMAL POWER, and L2

~~c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.~~ L3

SR3.2.2.2 4.2.3.2 The applicable MCPR limit shall be determined from the COLR based on:

a. Technical Specification Scram Speed (TSSS) MCPR limits, or

b. Nominal Scram Speed (NSS) MCPR limits if scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times identified in the COLR. LA2

Within 72 hours of completion of each set of scram testing, the results will be ~~compared against the nominal scram speed (NSS) insertion times specified in the COLR, to verify the applicability of the transient analyses.~~ Prior to initial scram time testing for an operating cycle, the MCPR operating limits used shall be based on the Technical Specification Scram Speeds (TSSS). LA3

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ITS 3.2.3

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE A.1

LIMITING CONDITION FOR OPERATION

LCO 3.2.3 3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. A.2

ACTION:

LA.1

ACTION A With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or

ACTION B reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 4.2.4 LHGR's shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER ≥25% increase of at least 15% of RATED THERMAL POWER, and L.1

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR. L.2

Page 2 of 2

ITS 3.3.1.1

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION ~~SYSTEM~~ INSTRUMENTATION A.1

LIMITING CONDITION FOR OPERATION

LCO 3.3.1.1 / SR 3.3.1.1.17 3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME ~~as shown in Table 3.3.1-2.~~ LA.1

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION: Add proposed ACTIONS NOTE 1 A.2

Action A a. With one channel required by Table 3.3.1-1 inoperable ~~in one or more Functional Units,~~ place the inoperable channel and/or that trip system in the tripped condition~~*~~ within 12 hours. LA.2

Actions A,B and C b. With two or more channels required by Table 3.3.1-1 inoperable in one or more Functional Units: LA.2

Action C 1. Within one hour, verify sufficient channels remain OPERABLE or tripped~~*~~ to maintain trip capability in the Functional Unit, and

Action B 2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system~~**~~ in the tripped condition~~*~~, and LA.2

Action A 3. Within 12 hours, restore the inoperable channels in the other trip system to an OPERABLE status or tripped~~*~~. LA.2

Action D c. Otherwise, take the ACTION required by Table 3.3.1-1 for the Functional Unit.

SURVEILLANCE REQUIREMENTS

Note 1 to Surveillance Requirements 4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

SR 3.3.1.1.15 4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS ~~and simulated automatic operation~~ LA.3 of all channels shall be performed at least once per ~~18~~ 24 LD.1 months.

SR 3.3.1.1.17 4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of ~~each~~ reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per ~~18~~ months. A.3 ~~Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.~~ Add proposed Note 4 A.3

~~* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur.~~ LA.2 In these

Action D cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.3.1-1 for the Functional Unit shall be taken.

~~** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.~~

Addressed by Definition of STAGGERED TEST BASIS, Note 3, and DOC A.5

A.1

Table 3.3.1.1-1

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

Note 2 to Surveillance Requirements

L.2 Add proposed Note (a) to Table 3.3.1.1-1

Function	FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1.a	1. Intermediate Range Monitors: a. Neutron Flux - High	2 ~~3, 4~~ 5~~(b)~~ LA.4	3 ~~2~~ 3	G 1 ~~2~~ L.1 H 3
1.b	b. Inoperative	2 ~~3, 4~~ 5	3 ~~2~~ 3	G 1 ~~2~~ L.1 H 3
2.a	2. Average Power Range Monitor:~~(c)~~ LA.5 a. Neutron Flux - High, Setdown	2 ~~3~~ ~~5(b)~~	2 ~~2~~ ~~2~~	G1 ~~2~~ L.3 ~~3~~ L.4
2.b	b. Flow Biased Simulated Thermal Power-Upscale	1	2	F 4
2.c	c. Fixed Neutron Flux-High	1	2	F 4
2.d	d. Inoperative	1, 2 ~~3~~ ~~5~~	2 ~~2~~ ~~2~~	G1 ~~2~~ L.3 ~~3~~ L.4
3.	3. Reactor Vessel Steam Dome Pressure - High	1, 2~~(d)~~ A.4	2	G 1
4.	4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	G1
5	5. Main Steam Line Isolation Valve - Closure	1~~(c)~~ LA.6	~~4~~ ⑧ A.5	F 4
	6. DELETED			

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Table 3.3.1.1-1

A.1

TABLE 3.3.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

{Note 2 to Surveillance Requirements

	Function FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM ~~(a)~~	ACTION	
6.	~~7.~~ Primary Containment Pressure - High	1, 2~~(j)~~ A.4	2~~(l)~~ LA.6	G 1	
7.a and 7.b	~~8.~~ Scram Discharge Volume Water Level - High	1, 2, 5(h) {Note (a) to Table 3.3.1.1-1	~~2~~ ~~2~~ (4) A.6	G 1 H 3	
8.	~~9.~~ Turbine Stop Valve - Closure	1~~(j)~~ LA.5	4~~(l)~~ LA.6	E 6	
9.	~~10.~~ Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1~~(j)~~ LA.5	2~~(l)~~ LA.6	E 6	
10.	~~11.~~ Reactor Mode Switch Shutdown Position	1, 2 ~~3, 4~~ 5	~~1~~ (2) A.5 ~~1~~ ~~1~~ (2) A.5	G 1 ~~7~~ H 3	L.1
11.	~~12.~~ Manual Scram	1, 2 ~~3, 4~~ 5	~~1~~ (2) M.2 ~~1~~ ~~1~~ (2) M.2	G 1 ~~8~~ H 9	L.1
	~~13. Control Rod Drive~~				R.1
	~~a. Charging Water Header Pressure - Low~~	~~2 5(h)~~	~~2 2~~	~~1 3~~	
	~~b. Delay Timer~~	~~2 5(h)~~	~~2 2~~	~~1 3~~	

L.2 — Add proposed Note (a) to Table 3.3.1.1-1

ITS 3.3.1.1

A.1

ITS 3.3.1.1

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION G { ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

~~ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.~~ L.1 L.3

ACTION H { ACTION 3 - ~~Suspend all operations involving CORE ALTERATIONS and~~ insert all insertable control rods ~~within one hour~~ Immediately Initiate action to fully L.2 A.8

in core cells containing one or more fuel assemblies. L.2 L.6

ACTION F { ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - DELETED

ACTION E { ACTION 6 - ~~Initiate a reduction in THERMAL POWER within 15 minutes and~~ reduce THERMAL POWER to less than 25% of RATED THERMAL POWER, within ~~2~~ 4 hours. L.7 L.8

~~ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.~~ L.1

ACTION 8 - ~~Lock the reactor mode switch in the Shutdown position within 1 hour.~~ L.2

ACTION H { ACTION 9 - ~~Suspend all operations involving CORE ALTERATIONS and~~ insert all insertable control rods ~~and lock the reactor mode switch in the SHUTDOWN position~~ ~~within 1 hour~~. L.2

Immediately A.8

in core cells containing one or more fuel assemblies. A.8

Initiate action to fully A.8

ITS 3.3.1.1

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION A.1

TABLE NOTATIONS

Note 2 to Surveillance Requirements

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the channel in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) ~~The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and during shutdown margin demonstrations performed per Specification 3.10.3.~~ LA.4

(c) ~~An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.~~ LA.5

(d) ~~This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.~~ A.4

(e) ~~This function shall be automatically bypassed when the reactor mode switch is not in the Run position.~~ LA.6

(f) ~~This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.~~ A.4

(g) ~~Also actuates the standby gas treatment system.~~ LA.6

Note (a) to Table 3.3.1-1

(h) With any control rod withdrawn from a control cell containing one or more fuel assemblies. ~~Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ A.7 L.6

(i) ~~This function shall not be automatically bypassed when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.~~ LA.5

(j) ~~Also actuates the EOC-RPT system.~~ LA.6

~~*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ LA.4

A.1

TABLE 3.3.1-2
REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor*	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power-Upscale	≤ 0.09**
c. Fixed Neutron Flux - High	≤ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55##
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05##
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. DELETED	
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA
13. Control Rod Drive	
a. Charging Water Header Pressure - Low	NA
b. Delay Timer	NA

L.11

LA.1

|

Note 1 to SR 3.3.1.1.17 *Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. LA.1

**Not including simulated thermal power time constant. L.11

Note 5 to SR 3.3.1.1.17 #Measured from start of turbine control valve fast closure.

Note 2 to SR 3.3.1.1.17 ##Sensor is eliminated from response time testing for the RPS circuits. Response time testing and conformance to the administrative limits for the remaining channel including trip unit and relay logic are required. LA.9

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A.1

Table 3.3.1.1-1

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function FUNCTIONAL UNIT	SR 3.3.1.1.1 CHANNEL CHECK	SR 3.3.1.1.4, SR 3.3.1.1.5, SR 3.3.1.1.9 CHANNEL FUNCTIONAL TEST	SR 3.3.1.1.10, SR 3.3.1.1.11, SR 3.3.1.1.13 CHANNEL CALIBRATION (a) Note 1 to SR 3.3.1.1.13 Note 1 to SR 3.3.1.1.11	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1.a 1. Intermediate Range Monitors a. Neutron Flux - High	S/U(b), S	A.9 S/U(c), W-4	R-13	2* 3*, 4, 5 Note to SR 3.3.1.1.4 Note 2 to SR 3.3.1.1.13 M.1
1.b b. Inoperative SR 3.3.1.1.6 SR 3.3.1.1.7	NA	W-4, -5	LE.1 R-13 NA	2*, 3*, 4, 5 L.1 add proposed Note (a) to Table 3.3.1.1-1 L.2
2.a 2. Average Power Range Monitor: (f) SR 3.3.1.1.8 a. Neutron Flux - High, Setdown SR 3.3.1.1.7	S/U(b), S ~~S~~	A.9 S/U(c), W-4 ~~W~~	SA-11 L.3 L.4 ~~SA~~	2* L.3 3*, 5 L.4 Note to SR 3.3.1.1.4 Note 2 to SR 3.3.1.1.11
2.b b. Flow Biased Simulated Thermal Power-Upscale	S, D(e) A.10	L.5 S/U(c), Q-9	W(d)(e) SR 3.3.1.1.3, SA-11, R(h) LE.1 SR 3.3.1.1.2	1
2.c c. Fixed Neutron Flux - High	S	S/U(c), Q-9	W(d), SA-11	1
2.d d. Inoperative	NA	Q-9	NA SR 3.3.1.1.14	1, 2, 3, 5 L.3 L.4
3. 3. Reactor Vessel Steam Dome Pressure - High	NA	Q-9	Q-10	1, 2
4. 4. Reactor Vessel Water Level - Low, Level 3	S	Q-9	LE.1 R-13	1, 2
5. 5. Main Steam Line Isolation Valve - Closure	NA	Q-9	LE.1 R-13	1
6. DELETED				
6. 7. Primary Containment Pressure - High	NA	Q-9	LE.1 Q-13	1, 2

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Table 3.3.1.1-1 A.1

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function	FUNCTIONAL UNIT	SR 3.3.1.1.1 CHANNEL CHECK	SR3.3.1.1.4; SR3.3.1.1.5; SR 3.3.1.1.9; SR 3.3.1.1.12 CHANNEL FUNCTIONAL TEST	SR 3.3.1.1.10; SR 3.3.1.1.11; SR 3.3.1.1.13 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED L.6
7.	8. Scram Discharge Volume Water Level - High	NA	Q-9	LE.1 R-13	1, 2, 5 ← add proposed Note (a) to Table 3.3.1.1-1
8.	9. Turbine Stop Valve - Closure (1)	NA	Q-9	R-13	1
9.	10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low (1) — SR 3.3.1.1.16	NA	Q-9	LE.1 R-13	1 L.1 add proposed Note (a) to Table 3.3.1.1-1
10.	11. Reactor Mode Switch Shutdown Position	NA	LD.2 R-12	NA	1, 2, ~~3, 4~~, 5
11.	12. Manual Scram	NA	W-~~5~~	NA	1, 2, ~~3, 4~~, 5 L.2
	~~13. Control Rod Drive~~				
	~~a. Charging Water Header Pressure - Low~~	~~NA~~	~~M~~	~~R~~	~~2, 5~~ R.T
	~~b. Delay Timer~~	~~NA~~	~~M~~	~~R~~	~~2, 5~~

Note 1 to SR 3.3.1.1.11
Note 1 to SR 3.3.1.1.13 — (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

SR3.3.1.1.6
SR 3.3.1.1.7 — (b) The IRM and SRM channels shall be determined to overlap ~~for at least 1/2 decades~~ during each startup and the IRM and APRM channels shall be determined to overlap ~~for at least 1/2 decades~~ during each controlled shutdown, if not performed within the previous 7 days. LA.3

~~(c) Within 24 hours prior to startup, if not performed within the previous 7 days.~~ A.9 L.5

SR3.3.1.1.2 — (d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. ~~The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.~~ LA.8

Add proposed Note to SR 3.3.1.1.2 L.9

Actions Note 2 — Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98. ~~Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.~~

Sr 3.3.1.1.3 — (e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal. L.10

SR 3.3.1.1.8 — (f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH). A.10

~~(g) Measure and compare core flow to rated core flow.~~ A.13

SR 3.3.1.1.14 — (h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

SR 3.3.1.1.16 — (i) At least once per ~~18~~ months, verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low Trip Functions are not bypassed when THERMAL POWER is ≥ 25% of RATED THERMAL POWER. Specification 4.0.2 applies to this ~~18~~ 24 month interval. LD.1

Note to SR 3.3.1.1.4
Note 2 to SR 3.3.1.1.11
Note 2 to SR 3.3.1.1.13 — * The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1. M.1

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A.1

ITS 3.3.6.1

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.6.1 — 3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3. LA.1

SR 3.3.6.1.6 — LA.2

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION: — add proposed ACTIONS Note — A.2

ACTIONS A and B

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value. LA.1

ACTION A

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System Requirement for one trip system, either

1. Place the inoperable channel(s) and/or trip system in the tripped condition* within

   a) 1 hour for trip functions without an OPERABLE channel, LB.1
   b) 12 hours for trip functions common to RPS Instrumentation, and
   c) 24 hours for trip functions not common to RPS Instrumentation,

or

ACTION C — 2. Take the ACTION required by Table 3.3.2-1.

ACTION B

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems,

1. Place at least one trip system** in the tripped condition*** within one hour, and

ACTION A — 2. a) Place the inoperable channel(s) in the remaining trip system in the tripped condition*** within

   1) 1 hour for trip functions without an OPERABLE channel. LB.1
   2) 12 hours for trip functions common to RPS Instrumentation, and
   3) 24 hours for trip functions not common to RPS Instrumentation.

   or

ACTION C — b) Take the ACTION required by Table 3.3.2-1.

---

* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. LA.3 In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken. LB.1

** If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition except when this would cause the Trip Function to occur.

*** An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. LA.3 In these cases, the inoperable channel shall be restored to OPERABLE status within 1 hour or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken. LB.1

ACTION B

ACTION C

ITS 3.3.6.2

INSTRUMENTATION A.1

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.6.2 3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 ~~and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3~~. LA.1 A.2

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION: add proposed ACTIONS Note A.3

ACTIONS A and B a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value. LA.1

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System Requirement for one trip system, either

ACTION A 1. Place the inoperable channel(s) and/or trip system in the tripped condition* within

a) 1 hour for trip functions without an OPERABLE channel, LB.1
b) 12 hours for trip functions common to RPS Instrumentation, and
c) 24 hours for trip functions not common to RPS Instrumentation,

or

ACTION C 2. Take the ACTION required by Table 3.3.2-1.

ACTION B c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems,

1. Place at least one trip system** in the tripped condition*** within one hour, and

ACTION A 2. a) Place the inoperable channel(s) in the remaining trip system in the tripped condition*** within

1) 1 hour for trip functions without an OPERABLE channel, LB.1
2) 12 hours for trip functions common to RPS Instrumentation, and
3) 24 hours for trip functions not common to RPS Instrumentation,

or

ACTION C b) Take the ACTION required by Table 3.3.2-1. LA.2

---

* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. ~~In these cases, the~~ inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken. LB.1

** If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition except when this would cause the Trip Function to occur. LA.2

*** An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the ACTION B inoperable channel shall be restored to OPERABLE status within 1 hour or ACTION C the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken. LB.1

A.1

ITS 3.3.6.1

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Note 1 to Surveillance Requirements

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

LA.4

SR3.3.6.1.5

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months. 24 LD.1

SR3.3.6.1.6

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system.

A.3

24 LD.1

addressed by definition of STAGGERED TEST BASIS and A.4

ITS 3.3.6.2

A.1

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Note 1 to Surveillance Requirements

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

LA.3

SR 3.3.6.2.4

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS ~~and simulated automatic operation~~ of all channels shall be performed at least once per ~~18~~ 24 months.

LD.1

~~4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system.~~

A.2

Table 3.3.6.1-1

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

	Function TRIP FUNCTION	VALUE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A.5	A. ~~AUTOMATIC INITIATION~~				
A.5 1,2	1. PRIMARY CONTAINMENT ISOLATION				
	a. Reactor Vessel Water Level				
2.f	(1) Low, Level 3	7	2	1, 2, 3	L.1 (20) F
2.a	(2) Low Low, Level 2	2, 3	2	1, 2, 3	20 H
1.a, 2.e	(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	(20) D (group 1) F (group 10) L.2 for group 1, L.1 group 10
2.b	b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20 H
	c. Main Steam Line				
	1) DELETED				
1.b	2) Pressure - Low	1	2	1	23 E
1.c	3) Flow - High	1	2/line(d) M.1	1, 2, 3	(21) D L.3
	d. DELETED				
1.e	e. Main Steam Line Tunnel ΔTemperature - High	1	2	1(1)(1), 2(1)(1), 3(1)(1),	(21) D
1.d	f. Condenser Vacuum - Low	1	2	1, 2(a), 3(a) - Footnote (a)	(21) D
2	2. SECONDARY CONTAINMENT ISOLATION A.6 moved to ITS 3.3.6.2				
2.c	a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24 F
2.b	b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24 H
2.a	c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24 H
2.d	d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24 F

LA.5

(b) Note 2 to Surveillance Requirements

M.1

Note 2 and 3 to Surveillance Requirements

L.1

L.2

L.3

LA.5

A.6 moved to ITS 3.3.6.2

M.2 add proposed ACTIONS F and H as indicated

A.1

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ITS 3.3.6.1

Table 3.3.6.2-1

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

LA.4

Function TRIP FUNCTION	~~VALVE GROUPS OPERATED BY SIGNAL~~	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. ~~AUTOMATIC INITIATION~~ A.4				<See ITS 3.3.6.1>
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) DELETED				
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line$^{(d)}$	1, 2, 3	21
d. DELETED				
e. Main Steam Line Tunnel ΔTemperature - High	1	2	$1^{(i)(j)}$, $2^{(i)(j)}$, $3^{(i)(j)}$	21
f. Condenser Vacuum - Low	1	2	1, 2^{*}, 3^{*}	21
2. SECONDARY CONTAINMENT ISOLATION				
3 a. Reactor Building Vent Exhaust Plenum Radiation - High	4~~$^{(c)(e)}$~~ LA.4	2	1, 2, 3 and ** (Notes (a) and (b) to Table 3.3.6.2-1)	24C
2 b. Drywell Pressure - High	4~~$^{(c)(e)}$~~	2	1, 2, 3	24C
1 c. Reactor Vessel Water Level - Low Low, Level 2	4~~$^{(c)(e)}$~~	2	1, 2, 3, and * (Note (a) to Table 3.3.6.2-1)	24C
4 d. Fuel Pool Vent Exhaust Radiation - High	4~~$^{(c)(e)}$~~	2	1, 2, 3, and ** (Notes (a) and (b) to Table 3.3.6.2-1)	24 C

<See ITS 3.3.6.1>

A.1

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ITS 3.3.6.2

Table 3.3.6.1-1

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

Function

	TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
4.	3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
4.a	a. Δ Flow - High	5	1	1, 2, 3	22 F
4.c	b. Heat Exchanger Area Temperature - High	5	1/heat exchanger	1, 2, 3	22 F
4.d	c. Heat Exchanger Area Ventilation ΔT - High	5	1/heat exchanger	1, 2, 3	22 F
4.l	d. SLCS Initiation	5 (1) A.9	(2) NA	1, 2, (3) L.4	(22) I L.11
4.k	e. Reactor Vessel Water Level - Low Low, Level 2	5	2	1, 2, 3	22 F
4.e	f. Pump and Valve Area Temperature - High	5	1/area	1, 2, 3	22 F
4.f	g. Pump and Valve Area Ventilation ΔT - High	5	1/area	1, 2, 3	22 F
4.g	h. Holdup Pipe Area Temperature - High	5	1	1, 2, 3	22 F
4.h	i. Holdup Pipe Area Ventilation ΔT - High	5	1	1, 2, 3	22 F
4.i	j. Filter/Demineralizer Valve Room Area Temperature - High	5	1	1, 2, 3	22 F
4.j	k. Filter/Demineralizer Valve Room Area Ventilation ΔT - High	5	1	1, 2, 3	22 F
	~~l. Pump Suction Flow - High~~	~~5~~	~~1~~	~~1, 2, 3~~	~~22~~ LA.9

add proposed Function 4.b

M.3

Note 2 to Surveillance Requirements

footnote (b)

LA.5

A.1

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ITS 3.3.6.1

Table 3.3.6.1-1

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

	Function TRIP FUNCTION	LA.5 ~~VALVE GROUPS OPERATED BY SIGNAL~~	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b) Note 2 to Surveillance Requirements	APPLICABLE OPERATIONAL CONDITION	ACTION
3	4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
3.a	a. RCIC Steam Line Flow - High	~~8~~	1	1, 2, 3	22 F
3.c	b. RCIC Steam Supply Pressure - Low	~~8, 9(g)~~	2	1, 2, 3	22 F
3.d	c. RCIC Turbine Exhaust Diaphragm Pressure - High	~~8~~	2	1, 2, 3	22 F
3.e	d. RCIC Equipment Room Temperature - High	~~8~~	1	1, 2, 3	22 F
3.g	e. RCIC Steam Line Tunnel Temperature - High	~~8~~	1	1, 2, 3	22 F
3.h	f. RCIC Steam Line Tunnel Δ Temperature - High	~~8~~	1	1, 2, 3	22 F
3.i	g. Drywell Pressure - High	~~9(g)~~	2	1, 2, 3	22 F
3.f	h. RCIC Equipment Room Δ Temperature - High	~~8~~	1	1, 2, 3	22 F

LA.5

A.1

add proposed Function 3.b

M.3

Page 22 of 34

ITS 3.3.6.1

Table 3.3.6.1-1

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

Note 2 to Surveillance Req'ments

LA.5

	Function TRIP FUNCTION	~~VALVE GROUPS OPERATED BY SIGNAL~~	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
	~~5 DELETED~~				
5.	6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		M4 add proposed Note (2)		
5.a	a. Reactor Vessel Water Level - Low, Level 3	LA.5 ~~6~~	2	L.6 ~~1/2~~, 3 M.4 4 and 5	L.8 25 J
5.b	b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	~~6~~	1	1, 2, 3	L7 25 F
	~~c. RHR Pump Suction Flow - High~~	~~6~~	~~1~~	~~1, 2, 3~~	~~25~~ LA.8
	A.5 ~~B. MANUAL INITIATION~~		(2) M.5 for group1		
1.f, 2.g, 4.m, 5.c	1. ~~Inboard Valves~~ LA.5	~~1, 2, 5, 6, 7~~ LA.5	① ~~/group~~	1, 2, 3	26 G
1.f, 2.g, 4.m, 5.c	2. ~~Outboard Valves~~	~~1, 2, 5, 6~~	① ~~/group~~	1, 2, 3	26 G
2.g	3. ~~Inboard Valves~~	~~4~~ (c)(e) A.6 moved to ITS 3.3.6.2	1 ~~/group~~	1, 2, 3 A.6 moved to ITS 3.3.6.2 and **,#	26 G
2.g	4. ~~Outboard Valves~~	~~4~~ (c)(e)	1 ~~/group~~	1, 2, 3 and **,#	26 G
L.10	~~5. Inboard Valves~~	~~3, 8, 9~~	~~1/valve~~	~~1, 2, 3~~	~~26~~
	~~6. Outboard Valves~~	~~3, 8, 9~~	~~1/valve~~	~~1, 2, 3~~	~~26~~
3.j	7. ~~Outboard Valve~~	~~8~~ (11) footnote(b) LA.5	1 ~~/group~~	1, 2, 3	26 G L.9

include Group 10 with ITS Function 2.g M.7

A6 moved to ITS 3.3.6.2

A.1

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ITS 3.3.6.1

Table 3.3.6.2-1

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

Function TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
	LA.4			
5. DELETED				
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	6	2	1, 2, 3	25
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	6	1	1, 2, 3	25
c. RHR Pump Suction Flow - High	6	1	1, 2, 3	25
5. B. MANUAL INITIATION				
1. Inboard Valves	1, 2, 5, 6, 7	1/group	1, 2, 3	26
2. Outboard Valves	1, 2, 5, 6	1/group	1, 2, 3	26
5. 3. Inboard Valves	$4^{(j)(m)}$	1/group	1, 2, 3 and **, #	26
5. 4. Outboard Valves	$4^{(k)(m)}$	1/group	1, 2, 3 and **, #	26
5. Inboard Valves	3, 8, 9	1/valve	1, 2, 3	26
6. Outboard Valves	3, 8, 9	1/valve	1, 2, 3	26
7. Outboard Valve	$8^{(m)}$	1/group	1, 2, 3	26

See ITS 3.3.6.1

LA.4

See ITS 3.3.6.1

LA.4

Notes (a) and (b)

A.1

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ITS 3.3.6.2

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

add proposed Required Action D.1 — L.2

ACTION STATEMENTS

ACTIONS D, F, and H — ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours. ← add proposed ACTION F — L.1

ACTION D — ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 12 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours. — L.3

add proposed Required Action I.1 — L.11

ACTIONS F and I — ACTION 22 - Close the affected system isolation valves within 1 hour ~~and declare the affected system inoperable~~. — A.8

ACTION E — ACTION 23 - Be in at least STARTUP within 6 hours.

A.6 moved to ITS 3.5.6.2

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

L.7 — close

ACTIONS F and J — ACTION 25 - Lock the affected system isolation valves closed within 1 hour and ~~declare the affected system inoperable~~. — L.8 — A.8 ← add proposed Required Actions J.1 and J.2

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise: — ACTION G — L.9

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or — ACTION H

b. Close the affected system isolation valves within the next hour ~~and declare the affected system inoperable~~. — ACTION G — A.8

TABLE NOTATIONS

Table 3.3.6.1-1 Footnote (a)
* May be bypassed with all turbine stop valves not full open.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

~~(a) Deleted~~.

Note 2 to Surveillance Requirement
(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition. — the associated Function maintains isolation capability — LB.2

A.6 moved to ITS 3.3.6.2
~~(c) Also actuates the standby gas treatment system.~~

~~(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.~~ — M.1

~~(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.~~

Table 3.3.6.1 footnote (b) (f) ~~Closes only RWCU system inlet outboard valve.~~ — LA.5

only inputs into one of two trip systems

A.1

ITS 3.3.6.2

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

L.1 — add proposed Required Actions C.1.2 and C.2.2

<See ITS 3.3.6.1>

ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

A.5

ACTION C — ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

<See ITS 3.3.6.1> ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise: M.1

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or A.9

ACTION C — b. Close the affected system isolation valves within the next hour and declare the affected system inoperable. L.2 — add proposed Required Action C.2.1

or L.4

<See ITS 3.3.6.1>

TABLE NOTATIONS

* May be bypassed with all turbine stop valves not full open.

Notes (a) and (b) to Table 3.3.6.2-1 — ** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

Note (a) to Table 3.3.6.2-1 — # During CORE ALTERATIONS and operations with a potential for draining the reactor vessel. L.3

(a) Deleted.

maintains isolation capability

Note 2 to Surveillance Requirements — (b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition. LB.2

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE. LA.4

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

<See ITS 3.3.6.1>

A.1

ITS 3.3.6.1

TABLE 3.3.2-1 (Continued)

NOTES (Continued)

LA.5

Table 3.3.6.1-1 footnote (b)

(g) Requires RCIC steam supply pressure-low coincident with drywell pressure-high.

(h) Manual initiation isolates 2E51-F008 only and only with a coincident reactor vessel water level-low, level 2, signal.

LA.5

Notes 2 and 3 to ACTIONS

(i) Both channels of each trip system may be placed in an inoperable status for up to 4 hours for required reactor building ventilation system corrective maintenance, filter changes, damper cycling and surveillance tests, other than Surveillance Requirement 4.6.5.1.c, without placing the trip system in the tripped condition.

(j) Both channels of each trip system may be placed in an inoperable status for up to 12 hours due to loss of reactor building ventilation or for performance of Surveillance Requirement 4.6.5.1.c without placing the trip system in the tripped condition.

Only inputs into one of two trip systems

Table 3.3.6.1-1

TABLE 3.3.2-2
ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

A.6
1, 2

Function TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
2.f 1) Low, Level 3	~~≥ 12.5 inches*~~	≥ 11.0 inches*
2.a 2) Low Low, Level 2	~~≥ -50 inches*~~	≥ -57 inches*
1.a, 2.e 3) Low Low Low, Level 1	~~≥ -129 inches*~~	≥ -136 inches*
2.b b. Drywell Pressure - High	~~≤ 1.69 psig~~	≤ 1.89 psig
c. Main Steam Line		
1) DELETED		
1.b 2) Pressure - Low	~~≥ 854 psig~~	≥ 834 psig
1.c 3) Flow - High	~~≤ 111 psid~~	≤ 116 psid
d. DELETED		
1.e e. Main Steam Line Tunnel Δ Temperature - High	~~≤ 65°F~~	≤ 70°F
1.d f. Condenser Vacuum - Low	~~> 7 inches Hg vacuum~~	≥ 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
2.c a. Reactor Building Vent Exhaust Plenum Radiation - High	~~≤ 10 mr/h~~	≤ 15 mr/h
2.b b. Drywell Pressure - High	~~≤ 1.69 psig~~	≤ 1.89 psig
2.a c. Reactor Vessel Water Level - Low Low, Level 2	~~≥ -50 inches*~~	≥ -57 inches*
2.d d. Fuel Pool Vent Exhaust Radiation - High	~~≤ 10 mr/h~~	≤ 15 mr/h
4 3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
4.a a. ΔFlow - High	~~≤ 70 gpm~~	≤ 87.5 gpm
4.c b. Heat Exchanger Area Temperature - High	~~≤ 149°F~~	≤ 156.8°F
4.d c. Heat Exchanger Area Ventilation ΔT - High	~~≤ 33°F~~	≤ 40.3°F
4.l d. SLCS Initiation	~~N.A.~~	N.A.
4.k e. Reactor Vessel Water Level - Low Low, Level 2	~~≥ -50 inches*~~	> -57 inches*

LA.1

LF.1

A.1

A.11

A.11

A11

A.6
moved to ITS 3.3.6.2

M.3
add proposed function 4.b

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ITS 3.3.6.1

Table 3.3.6.2-1

TABLE 3.3.2-2
ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

Function TRIP FUNCTION	~~TRIP SETPOINT~~ LA.1	ALLOWABLE VALUE
A.4 A. ~~AUTOMATIC INITIATION~~		
<see ITS 3.3.6.1> 1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) DELETED		
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 111 psid	≤ 116 psid
d. DELETED		
e. Main Steam Line Tunnel Δ Temperature - High	≤ 65°F	≤ 70°F
f. Condenser Vacuum - Low	> 7 inches Hg vacuum	> 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION	LA.1	LF.1
3 a. Reactor Building Vent Exhaust Plenum Radiation - High	~~≤ 10 mr/h~~	≤ 15 mr/h
2 b. Drywell Pressure - High	~~≤ 1.69 psig~~	≤ 1.89 psig
1 c. Reactor Vessel Water Level - Low Low, Level 2	~~≥ -50 inches*~~	≥ -57 inches~~*~~ A.6
4 d. Fuel Pool Vent Exhaust Radiation - High	~~≤ 10 mr/h~~	≤ 15 mr/h
<see ITS 3.3.6.1> ~~3. REACTOR WATER CLEANUP SYSTEM ISOLATION~~		
~~a. ΔFlow - High~~	~~≤ 70 gpm~~	~~≤ 87.5 gpm~~
~~b. Heat Exchanger Area Temperature - High~~	~~≤ 149°F~~	~~≤ 156.8°F~~
~~c. Heat Exchanger Area Ventilation ΔT - High~~	~~≤ 33°F~~	~~≤ 40.3°F~~
~~d. SLCS Initiation~~	~~N.A.~~	~~N.A.~~
~~e. Reactor Vessel Water Level - Low Low, Level 2~~	~~> -50 inches*~~	~~> -57 inches*~~

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ITS 3.3.6.2

Table 3.3.6.1-1

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

		Function TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4.e	f.	Pump and Valve Area Temperature - High	~~≤ 201°F~~	≤ 209°F
4.f	g.	Pump and Valve Area Ventilation ΔT - High	~~≤ 86°F~~	≤ 92.5°F
4.g	h.	Holdup Pipe Area Temperature - High	~~≤ 201°F~~	≤ 209°F
4.h	i.	Holdup Pipe Area Ventilation ΔT - High	~~≤ 86°F~~	≤ 92.5°F
4.i	j.	Filter/Demineralizer Valve Room Area Temperature - High	~~≤ 201°F~~	≤ 209°F
4.j	k.	Filter/Demineralizer Valve Room Area Ventilation ΔT - High	~~≤ 86°F~~	≤ 92.5°F
	~~l.~~	~~Pump Suction Flow - High~~	~~≤ 560 gpm~~	~~≤ 610 gpm~~

LA.1

LF.1

A.1

LA.9

Table 3.3.6.1-1

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

Function TRIP FUNCTION			TRIP ~~SETPOINT~~ LA.1	ALLOWABLE VALUE LF.1
3	4.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
3.a	a.	RCIC Steam Line Flow - High	~~≤ 290% of rated flow, 178" H_2O~~	≤ ~~295% of rated flow~~ 185" H_2O A.7
3.c	b.	RCIC Steam Supply Pressure - Low	~~≥ 57 psig~~	≥ 53 psig
3.d	c.	RCIC Turbine Exhaust Diaphragm Pressure - High	~~≤ 10.0 psig~~	≤ 20.0 psig
3.e	d.	RCIC Equipment Room Temperature - High	~~≤ 200°F~~	≤ 206°F
3.g	e.	RCIC Steam Line Tunnel Temperature - High	~~≤ 200°F~~	≤ 206°F
3.h	f.	RCIC Steam Line Tunnel Δ Temperature - High	~~≤ 117°F~~	≤ 123°F
3.i	g.	Drywell Pressure - High	~~≤ 1.69 psig~~	≤ 1.89 psig
3.f	h.	RCIC Equipment Room ΔT Temperature - High	~~≤ 120°F~~	≤ 126°F

5. DELETED

add proposed Function 3.b M.3

A.1

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ITS 3.3.6.1

Table 3.3.6.1-1

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

Function TRIP FUNCTION	~~TRIP SETPOINT~~	ALLOWABLE VALUE
5 6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
5.a a. Reactor Vessel Water Level - Low, Level 3	~~≥ 12.5 inches*~~ LA.1	≥ 11.0 inches~~*~~ A.11
5.b b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	~~≤ 135 psig**~~	≤ 145 psig~~**~~ LA.6
~~c. RHR Pump Suction Flow - High~~	~~≤ 180" H_2O~~	~~≤ 186" H_2O~~ LA.8
A.5 ~~B. MANUAL INITIATION~~	~~N.A.~~ LA.1	N.A.
5.c, 1.f, 2.g, 4.m 1. Inboard Valves		
2. Outboard Valves		
2g 3. Inboard Valves		
4. Outboard Valves		
L.10 5. Inboard Valves		
6. Outboard Valves		
3.j 7. Outboard Valve		

A.1

A.11 ~~*See Bases Figure B 3/4 3-1~~

LA.6 ~~**Corrected for cold water head with reactor vessel flooded~~

N.A. - Not Applicable.

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ITS 3.3.6.1

Table 3.3.6.2-1

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

Function TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 135 psig**	≤ 145 psig**
c. RHR Pump Suction Flow - High	≤ 180" H_2O	≤ 186" H_2O
5. B. MANUAL INITIATION	N.A.	N.A.
1. Inboard Valves		
2. Outboard Valves		
5. 3. Inboard Valves		
5 4. Outboard Valves		
5. Inboard Valves		
6. Outboard Valves		
7. Outboard Valve		

<see ITS 3.3.6.1>

LA.4

A.6

~~*See Bases Figure B 3/4 3-1.~~

**Corrected for cold water head with reactor vessel flooded. <see ITS 3.3.6.1>

N.A. - Not Applicable.

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ITS 3.3.6.2

ITS 3.3.6.1

A.1

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)# — Note to SR 3.3.6.1.6
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	N/A
2) Low Low, Level 2	N/A
3) Low Low Low, Level 1	≤ 1.0*#
b. Drywell Pressure - High	N/A
c. Main Steam Line	
1) DELETED	
2) Pressure - Low	≤ 2.0*#
3) Flow - High	≤ 0.5*#
d. DELETED	
e. Condenser Vacuum - Low	N/A
f. Main Steam Line Tunnel ΔTemperature - High	N/A
2. SECONDARY CONTAINMENT ISOLATION	N/A
a. Reactor Building Vent Exhaust Plenum Radiation - High	
b. Drywell Pressure - High	
c. Reactor Vessel Water Level - Low, Level 2	
d. Fuel Pool Vent Exhaust Radiation - High	
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	N/A
a. ΔFlow - High	
b. Heat Exchanger Area Temperature - High	
c. Heat Exchanger Area Ventilation ΔT-High	
d. SLCS Initiation	
e. Reactor Vessel Water Level - Low Low, Level 2	
f. Pump and Valve Area Temperature - High	
g. Pump and Valve Area Ventilation ΔT - High	
h. Holdup Pipe Area Temperature - High	
i. Holdup Pipe Area Ventilation ΔT - High	

ITS 3.3.6.2

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)$^{\#}$
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	N/A
2) Low Low, Level 2	N/A
3) Low Low Low, Level 1	≤ 1.0$^{*,\#}$
b. Drywell Pressure - High	N/A
c. Main Steam Line	
1) DELETED	
2) Pressure - Low	≤ 2.0$^{*,\#}$
3) Flow - High	≤ 0.5$^{*,\#}$
d. DELETED	
e. Condenser Vacuum - Low	N/A
f. Main Steam Line Tunnel ΔTemperature - High	N/A
2. SECONDARY CONTAINMENT ISOLATION	N/A
a. Reactor Building Vent Exhaust Plenum Radiation - High	
b. Drywell Pressure - High	
c. Reactor Vessel Water Level - Low, Level 2	
d. Fuel Pool Vent Exhaust Radiation - High	
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	N/A
a. ΔFlow - High	
b. Heat Exchanger Area Temperature - High	
c. Heat Exchanger Area Ventilation ΔT-High	
d. SLCS Initiation	
e. Reactor Vessel Water Level - Low Low, Level 2	
f. Pump and Valve Area Temperature - High	
g. Pump and Valve Area Ventilation ΔT - High	
h. Holdup Pipe Area Temperature - High	
i. Holdup Pipe Area Ventilation ΔT - High	

A.2

<See ITS 3.3.6.1>

ITS 3.3.6.1

A.1

~~TABLE 3.3.2-3 (Continued)~~

~~ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME~~

~~TRIP FUNCTION~~	~~RESPONSE TIME (Seconds)#~~
~~j. Filter/Demineralizer Valve Room Area Temperature - High~~	
~~k. Filter/Demineralizer Valve Room Area Ventilation ΔT - High~~	
~~l. Pump Suction Flow - High~~	
~~4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION~~	~~N/A~~
~~a. RCIC Steam Line Flow - High~~	
~~b. RCIC Steam Supply Pressure - Low~~	
~~c. RCIC Turbine Exhaust Diaphragm Pressure - High~~	
~~d. RCIC Equipment Room Temperature - High~~	
~~e. RCIC Steam Line Tunnel Temperature - High~~	
~~f. RCIC Steam Line Tunnel ΔTemperature - High~~	
~~g. Drywell Pressure - High~~	
~~h. RCIC Equipment Room ΔTemperature - High~~	
~~5. DELETED~~	
~~6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION~~	~~N/A~~
~~a. Reactor Vessel Water Level - Low, Level 3~~	
~~b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High~~	
~~c. RHR Pump Suction Flow - High~~	
~~B. MANUAL INITIATION~~	~~N/A~~
~~1. Inboard Valves~~	
~~2. Outboard Valves~~	
~~3. Inboard Valves~~	
~~4. Outboard Valves~~	
~~5. Inboard Valves~~	
~~6. Outboard Valves~~	
~~7. Outboard Valve~~	

LA.2

L.10

TABLE NOTATIONS

SR 3.3.6.1.6 * Isolation system instrumentation response time for MSIVs only. ~~No diesel generator delays assumed.~~ LA.2

~~# Isolation system instrumentation response time specified for the Trip Function actuating the MSIVs shall be added to MSIV isolation time to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.~~ LA.2

Note to SR 3.3.6.1.6 ## Sensor is eliminated from response time testing for the MSIV actuation logic circuits. ~~Response time testing and conformance to the administrative limits for the remaining channel including trip unit and relay logic are required.~~ LA.7

~~N/A Not Applicable.~~ LA.2

Table 3.3.6.1-1

TABLE 3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

A.1

A.5

Function	TRIP FUNCTION	SR 3.3.6.1.1 CHANNEL CHECK	SR 3.3.6.1.2 CHANNEL FUNCTIONAL TEST	SR 3.3.6.1.3 SR 3.3.6.1.4 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
	A. AUTOMATIC INITIATION				
1, 2	1. PRIMARY CONTAINMENT ISOLATION				
	a. Reactor Vessel Water Level				
2.f	1) Low, Level 3	S	Q	4 - Ⓡ 24 months LE.1	1, 2, 3
2.a	2) Low Low, Level 2	NA	Q	4 - Ⓡ	1, 2, 3
1.a, 2.e	3) Low Low Low, Level 1	S	Q	4 - Ⓡ	1, 2, 3
2.b	b. Drywell Pressure - High	NA	Q	4 - Ⓡ	1, 2, 3
	c. Main Steam Line				
	1) DELETED				
1.b	2) Pressure - Low	NA	Q	3 - Q	1
1.c	3) Flow - High	NA	Q	3 - Ⓡ 92 days M.6	1, 2, 3
	d. DELETED				
1.d	e. Condenser Vacuum - Low	NA	Q	4 - Ⓡ	1, 2*, 3*
1.e	f. Main Steam Line Tunnel Δ Temperature - High	NA	Q	4 - Ⓡ 24 months LE.1	1, 2, 3
A.6 moved to ITS 3.3.6.2	2. SECONDARY CONTAINMENT ISOLATION				
2.c	a. Reactor Building Vent Exhaust Plenum Radiation - High	S	Q	4 - Ⓡ	1, 2, 3 and **
2.b	b. Drywell Pressure - High	NA	Q	4 - Ⓡ	1, 2, 3
2.a	c. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	4 - Ⓡ	1, 2, 3, and #
2.d	d. Fuel Pool Vent Exhaust Radiation - High	S	Q	4 - Ⓡ 24 months LE.1	1, 2, 3 and ** A.6 moved to ITS 3.3.6.2
4	3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
4.a	a. Δ Flow - High	S	Q	4 - Ⓡ	1, 2, 3
4.c	b. Heat Exchanger Area Temperature - High	NA	Q	4 - Ⓡ	1, 2, 3
4.d	c. Heat Exchanger Area Ventilation ΔT - High	NA	Q	4 - Ⓡ	1, 2, 3
4.l	d. SLCS Initiation	NA	Ⓡ A.12 24 months LD.1	NA	1, 2, ③ L.4
4.k	e. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	4 - Ⓡ 24 months LE.1	1, 2, 3

M.3 add proposed Function 4.b

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ITS 3.3.6.1

Table 3.3.6.2-1

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

A.4

Function TRIP FUNCTION	SR 3.3.6.2.1 CHANNEL CHECK	SR 3.3.6.2.2 CHANNEL FUNCTIONAL TEST	SR 3.3.6.2.3 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	S	Q	R	1, 2, 3
2) Low Low, Level 2	NA	Q	R	1, 2, 3
3) Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Main Steam Line				
1) DELETED				
2) Pressure - Low	NA	Q	Q	1
3) Flow - High	NA	Q	R	1, 2, 3
d. DELETED				
e. Condenser Vacuum - Low	NA	Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
3 a. Reactor Building Vent Exhaust Plenum Radiation - High	S	Q	~~R~~ 24 months	1, 2, 3 and **
2 b. Drywell Pressure - High	NA	Q	~~R~~ 24 months	1, 2, 3
1 c. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	~~R~~ 24 months	1, 2, 3, and #
4 d. Fuel Pool Vent Exhaust Radiation - High	S	Q	~~R~~ 24 months	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	Q	Q	1, 2, 3
d. SLCS Initiation	NA	Q	Q	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	R	NA	1, 2, 3
	NA	Q	R	1, 2, 3

<See ITS 3.3.6.1>

<See ITS 3.3.6.1>

LE.1

Notes (a) and (b) to Table 3.3.6.2-1

Note (a) to Table 3.3.6.2-1

Notes (a) and (b) to Table 3.3.6.2-1

A.1

ITS 3.3.6.2

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3.3.6.1-1

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function	TRIP FUNCTION	SR3.3.6.1.1 CHANNEL CHECK	SR 3.3.6.1.2 CHANNEL FUNCTIONAL TEST	SR 3.3.6.1.3 SR 3.3.6.1.4 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
4.e	f. Pump and Valve Area Temperature - High	NA	Q	4-Ⓠ	1, 2, 3
4.f	g. Pump and Valve Area Ventilation ΔT - High	NA	Q	4-Ⓠ	1, 2, 3
4.g	h. Holdup Pipe Area Temperature - High	NA	Q	4-Ⓠ	1, 2, 3
4.h	i. Holdup Pipe Area Ventilation ΔT - High	NA	Q	4-Ⓠ	1, 2, 3
4.i	j. Filter/Demineralizer Valve Room Area Temperature - High	NA	Q	4-Ⓠ	1, 2, 3
4.j	k. Filter/Demineralizer Valve Room Area Ventilation ΔT - High	NA	Q	4-Ⓠ	1, 2, 3
	~~l. Pump Suction Flow - High~~	~~S~~	~~Q~~	~~R~~	~~1, 2, 3~~
3	4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
3.a	a. RCIC Steam Line Flow - High	NA	Q	Q - 3	1 2, 3
3.b	b. RCIC Steam Supply Pressure - Low	NA	Q	4-Ⓠ	1 2, 3
3.d	c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	Q	4-Ⓠ	1 2, 3
3.e	d. RCIC Equipment Room Temperature - High	NA	Q	4-Ⓠ	1 2, 3
3.g	e. RCIC Steam Line Tunnel Temperature - High	NA	Q	4-Ⓠ	1 2, 3
3.h	f. RCIC Steam Line Tunnel Δ Temperature - High	NA	Q	4-Ⓠ	1 2, 3
3.i	g. Drywell Pressure - High	NA	Q	4-Ⓠ	1 2, 3
3.f	h. RCIC Equipment Room Δ Temperature - High	NA	Q	4-Ⓠ	1 2, 3

~~5. DELETED~~

24 months

LE.1

LA.9

A.1

M.3

add proposed Function 3.b

24 months

LE.1

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ITS 3.3.6.1

3.3.6.1-1

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Function TRIP FUNCTION	SR 3.3.6.1.1 CHANNEL CHECK	SR 3.3.6.1.2 CHANNEL FUNCTIONAL TEST	SR 3.3.6.1.3 SR 3.3.6.1.4 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED	
5	6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION					
5.a	a. Reactor Vessel Water Level - Low, Level 3	S	Q	4 R 24 months LE.1	1, 2 3 (4,5) L.6 M.4	
5.b	b. Reactor Vessel (RHR Cut-in Permissive) Pressure-High	NA	Q	4 Q	1, 2, 3	
	~~c. RHR Pump Suction Flow-High~~	~~NA~~	~~Q~~	~~Q~~	~~1, 2, 3~~	LA.8
A.5	~~B. MANUAL INITIATION~~					D.1
5.c, 1.f, 2.g, 4.m	1. Inboard Valves	NA	R A.12 24 months LD.1	NA	1, 2, 3	
5.c, 1.f, 2.g, 4.m	2. Outboard Valves	NA	R	NA	1, 2, 3	
2.g	3. Inboard Valves	NA	R	NA	1, 2, 3 ~~and **,#~~	A.6 moved to ITS 3.3.6.2
2.g	4. Outboard Valves	NA	R	NA	1, 2, 3 ~~and **,#~~	
L.10	~~5. Inboard Valves~~	~~NA~~	~~R~~	~~NA~~	~~1, 2, 3~~	
	~~6. Outboard Valves~~	~~NA~~	~~R~~	~~NA~~	~~1, 2, 3~~	
3.j	7. Outboard Valve	NA	R A.12 24 months LD.1	NA	1, 2, 3	

Footnote (a)

* Not required when all turbine stop valves are not full open.

~~** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.~~

~~# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.~~

A.6 moved to ITS 3.3.6.2

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ITS 3.3.6.1

Table 3.3.6.2-1

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function TRIP FUNCTION	SR 3.3.6.2.1 CHANNEL CHECK	SR 3.3.6.2.2 CHANNEL FUNCTIONAL TEST	SR 3.3.6.2.3 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION ⟨See ITS 3.3.6.1⟩				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure-High	NA	Q	Q	1, 2, 3
c. RHR Pump Suction Flow-High	NA	Q	Q	1, 2, 3
5. B. MANUAL INITIATION				
1. Inboard Valves	NA	R	NA	1, 2, 3
2. Outboard Valves	NA	R	NA	1, 2, 3
5. 3. ~~Inboard Valves~~ LA.4	A.7 NA	R* 24 months	NA LD.1	1, 2, 3 and **,#
5. 4. ~~Outboard Valves~~	NA	R*	NA	1, 2, 3 and **,#
5. Inboard Valves	NA	R	NA	1, 2, 3
6. Outboard Valves	NA	R	NA	1, 2, 3
7. Outboard Valve	NA	R	NA	1, 2, 3

A.1 Notes (a) and (b) to Table 3.3.6.2-1

⟨see ITS 3.3.6.1⟩

* Not required when all turbine stop valves are not full open.

Notes (a) and (b) to Table 3.3.6.2-1
** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

Notes (a) to Table 3.3.6.2-1
\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

L.3

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ITS 3.3.6.2

ITS 3.3.5.1

A.1

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LA.1

LCO 3.5.1 3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

A.2 moved to ITS 3.5.1 and 3.5.2

APPLICABILITY: As shown in Table 3.3.3-1. add proposed ACTIONS Note A.3

ACTION:

ACTION A a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value. LA.1

ACTION A b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1. L.1

ACTIONS E and F c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within: place channel in trip or

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.
2. 72 hours.

ACTION G Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours. A.4

150 L.2

SURVEILLANCE REQUIREMENTS

Note 1 to Surveillance Requirements 4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

LA.2

SR 3.3.5.1.5 4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months. 24 LD.1

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

A.2 moved to ITS 3.5.1 and 3.5.2

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A.1

ITS 3.3.8.1

INSTRUMENTATION

LOP

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

A.2

LIMITING CONDITION FOR OPERATION

LOP

LA.1

LCO 3.3.8.1 3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.~~ A.3

APPLICABILITY: As shown in Table 3.3.3-1. See proposed Actions Note A.4

ACTION:

LOP A.2

ACTION A

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status ~~with its trip setpoint adjusted consistent with the Trip Setpoint value.~~ LA.1

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1. LOP A.2

<SEE ITS 3.3.5.1>

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.
2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

A.2

LOP

Note 1 to Surveillances 4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1. (See ITS 3.3.5.1)

LA.2

SR 3.3.8.1.3 4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS ~~and simulated automatic operation~~ of all channels shall be performed at least once per ~~18~~ 24 months. LD.1

~~4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.~~

A.3

ITS 3.5.1

A.1

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 ~~and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3~~. LA.4

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.
2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

(See ITS 3.3.5.1)

SR 3.5.1.9 4.3.3.3* The ECCS RESPONSE TIME of each ECCS trip function ~~shown in Table 3.3.3-3~~ shall be demonstrated to be within the limit at least once per 18 24 months. ~~Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.~~

LA.4

LD.1

ADD proposed SR 3.5.1.9 Note A.5

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A.1

ITS 3.5.2

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 ~~and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3~~.

LA.4

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.
2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

LA.4

SR 3.5.2.7

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function ~~shown in Table 3.3.3-3~~ shall be demonstrated to be within the limit at least once per ~~18~~ 24 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

LD.1

(See ITS 3.3.5.1)

A.9 ADD proposed NOTE TO SR 3.5.2.7

Table 3.3.5.1-1

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

A.1

		Function TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION$^{(a)}$	APPLICABLE OPERATIONAL CONDITIONS	ACTION
	A.	~~DIVISION I TRIP SYSTEM~~ LA.3			
1.	1.	RHR-A (LPCI MODE) & LPCS SYSTEM			
a.	a.	Reactor Vessel Water Level - Low Low Low, Level 1	$2^{(b)}$	1, 2, 3, 4*, 5*	30 B
b.	b.	Drywell Pressure - High	$2^{(b)}$	1, 2, 3	30 B
c.	c.	LPCS Pump Discharge Flow-Low (Bypass)	1	1, 2, 3, 4*, 5*	31 D
g.	d.	LPCS and LPCI A Injection Valve Injection Line Pressure-Low (Permissive)	1/Valve	1, 2, 3 4*, 5*	32 C 33 B
d.	e.	LPCS and LPCI A Injection Valve Reactor Pressure-Low (Permissive)	2	1, 2, 3 4*, 5*	38 D 33 B
c.	f.	LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32 C
f.	g.	LPCI Pump A Discharge Flow-Low (Bypass)	1	1, 2, 3, 4*, 5*	31 D
h	h.	Manual Initiation	1/~~division~~ LA.3	1, 2, 3, 4*, 5*	34 C
4	2.	AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"① Note (c) to Table 3.3.5.1-1			
a.	a.	Reactor Vessel Water Level - Low Low Low, Level 1 ~~coincident with~~ LA.3	$2^{(b)}$	1, 2, 3	30 E
b.	b.	Drywell Pressure - High	$2^{(b)}$	1, 2, 3	30 E
c.	c.	Initiation Timer	1	1, 2, 3	32 F
d.	d.	Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32 E
e.	e.	LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	32 F
f.	f.	LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	32 F
h.	g.	Manual Initiation	M.6 ② ~~1~~/~~division~~ LA.3	1, 2, 3	34 F
g.	h.	Drywell Pressure Bypass Timer	M.1 ② ~~1~~	1, 2, 3	32 F
	~~i.~~	~~Manual Inhibit~~	~~1/division~~	~~1, 2, 3~~	~~34~~ R.1

ITS 3.3.5.1

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Table 3.3.5.1-1

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

A.1

		Function TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION$^{(a)}$	APPLICABLE OPERATIONAL CONDITIONS	ACTION
	B.	DIVISION 2 TRIP SYSTEM LA.3			
2.	1.	RHR B & C (LPCI MODE)			
a.	a.	Reactor Vessel Water Level - Low, Low Low, Level 1	$2^{(b)}$	1, 2, 3, 4*, 5*	30 B
b.	b.	Drywell Pressure - High	$2^{(b)}$	1, 2, 3	30 B
f.	c.	LPCI B and C Injection Valve Injection Line Pressure-Low (Permissive)	1/valve	1, 2, 3 4*, 5*	32 C 33 B
c.	d.	LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32 C
e.	e.	LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	31 D
g.	f.	Manual Initiation	1/division LA.3	1, 2, 3, 4*, 5*	34 C
d.	g.	LPCI B and C Injection Valve Reactor Pressure-Low (Permissive)	2	1, 2, 3, 4*, 5*	38 D 33 B
5	2.	AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B" ① - Note (c) to Table 3.3.5.1-1			
a.	a.	Reactor Vessel Water Level - Low Low Low, Level 1 coincident with	$2^{(b)}$	1, 2, 3	30 E
b.	b.	Drywell Pressure - High	$2^{(b)}$	1, 2, 3	30 E
c	c.	Initiation Timer	1	1, 2, 3	32 F
d.	d.	Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32 E
e.	e.	LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	32 F
g.	f.	Manual Initiation	M.6 ② ~~1~~/division LA.3	1, 2, 3	34 F
f.	g.	Drywell Pressure Bypass Timer	M.1 ② ~~1~~	1, 2, 3	32 F
	~~h.~~	~~Manual Inhibit~~	~~1/division~~	~~1, 2, 3~~	~~34~~

R.1

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ITS 3.3.5.1

Table 3.3.5.1-1

TABLE 3.3.3-1 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Function TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. ~~DIVISION 3 TRIP SYSTEM~~ — LA.3			
3 1. HPCS SYSTEM			
a. a. Reactor Vessel Water Level - Low, Low, Level 2	4[(b)]	1, 2, 3, 4*, 5*	35B
b. b. Drywell Pressure - High	4[(b)]	1, 2, 3	35B
c. c. Reactor Vessel Water Level-High, Level 8	2(c) — LA.3	1, 2, 3, 4*, 5*	32C
~~d. Deleted~~			
~~e. Deleted~~			
d. f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	31D
e. g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31D
f. h. Manual Initiation	1/division — LA.3	1, 2, 3, 4*, 5*	34C

D. LOSS OF POWER	TOTAL NO. OF INSTRUMENTS	INSTRUMENTS TO TRIP	MINIMUM OPERABLE INSTRUMENTS(d)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37

A.6 moved to ITS 3.3.8.1

A.1

TABLE NOTATION

Note 2 to Surveillance Requirements — (a) A channel/instrument may be placed in an inoperable status for up to 6 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter. — LB.1

Note (b) to Table 3.3.5.1-1 — (b) Also actuates the associated division diesel generator.

~~(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.~~ — LA.3

(d) A channel/instrument may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.

Note (a) to Table 3.3.5.1-1 — * Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

Note (c) to Table 3.3.5.1-1 — # Not required to be OPERABLE when reactor steam dome pressure is ≤ ~~122~~ 150 psig. — L.2

A.6 moved to ITS 3.3.8.1

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ITS 3.3.5.1

LOP A.2

Table 3.3.8.1-1

~~TABLE 3.3.3-1~~ (Continued)

~~EMERGENCY CORE COOLING SYSTEM ACTUATION~~ INSTRUMENTATION

FUNCTION

~~TRIP FUNCTION~~		MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C.	**DIVISION 3 TRIP SYSTEM**			
	1. **HPCS SYSTEM**			
<SEE ITS 3.3.5.1>	a. Reactor Vessel Water Level - Low, Low, Level 2	4[(b)]	1, 2, 3, 4*, 5*	35
	b. Drywell Pressure - High	4[(b)]	1, 2, 3	35
	c. Reactor Vessel Water Level-High, Level 8	2[(c)]	1, 2, 3, 4*, 5*	32
	d. Deleted			
	e. Deleted			
	f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	31
	g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31
	h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34

D. **LOSS OF POWER**	LA.3 ~~TOTAL NO. OF INSTRUMENTS~~	~~INSTRUMENTS TO TRIP~~	MINIMUM OPERABLE INSTRUMENTS(d)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1.a, 2.a, 2.b } 1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	~~2/bus~~	~~2/bus~~	2/bus	1, 2, 3, 4**, 5**	37
1.b, 1.c, 1.d, 2.c, 2.d, 2.e } 2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	~~2/bus~~	~~2/bus~~	2/bus	1, 2, 3, 4**, 5**	37

M.1

A and B

TABLE NOTATION

<SEE ITS 3.3.5.1>

(a) A channel/instrument may be placed in an inoperable status for up to 6 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.

(b) Also actuates the associated division diesel generator.

(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.

NOTE 2 TO SURVEILLANCES

(d) A channel/instrument may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition ~~provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter~~.

<SEE ITS 3.3.5.1>

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

APPLICABILITY

** Required when ESF equipment is required to be OPERABLE.

<SEE ITS 3.3.5.1>

Not required to be OPERABLE when reactor steam dome pressure is ≤ 122 psig.

M.2

Provided the associated Function maintains LOP initiation capability

ITS 3.3.8.1

A.1

Page 9 of 14

A.1

ITS 3.3.5.1

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
ACTION

ACTIONS B and E

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the ~~associated system inoperable.~~ A.7

ACTION G

add proposed Required Action B.1, B.3, E.1 and E.2

b. With more than one channel inoperable, declare the ~~associated system inoperable.~~ A.7 L.3

add proposed Required Action D.1 M.2

ACTION D

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function requirement, place ~~the inoperable channel in the tripped condition within 24 hours,~~ L.6 restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION G

M.2 add proposed Required Action C.1, E.1, F.1

ACTIONS C, E, and F

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare ~~the associated ADS trip system or~~ ECCS inoperable within 24 hours. A.7

L.1 for ADS Level 3 permissive

ACTION B

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 24 hours.

ACTIONS C and F

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or ~~declare the associated ADS trip system or~~ ECCS inoperable. A.7

ACTION B

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement. A.8

add proposed Required Action B.2

Channel

a. For ~~one trip system~~, place ~~that trip system~~ in the tripped condition within 24 hours or ~~declare the HPCS system inoperable.~~

add proposed Required Action B.2 and B.3

b. ~~For both trip systems, declare the HPCS system inoperable.~~ L.3

~~ACTION 36 - Deleted~~

ACTION 37 - With the number of OPERABLE instruments less than the Minimum Operable Instruments, place the inoperable instrument(s) in the tripped condition within 1 hour or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as appropriate.

A.6 moved to ITS 3.3.8.1

A.2 LOP

A.1

ITS 3.3.8.1

TABLE 3.3.3-1 (Continued)

~~EMERGENCY CORE COOLING SYSTEM ACTUATION~~ INSTRUMENTATION

ACTION

See ITS 3.3.5.1

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function requirement, place the inoperable channel in the tripped condition within 24 hours; restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable within 24 hours.

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 24 hours.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement

a. For one trip system, place that trip system in the tripped condition within 24 hours or declare the HPCS system inoperable.

b. For both trip systems, declare the HPCS system inoperable.

ACTION 36 - Deleted

ACTION 37 - With the number of OPERABLE instruments less than the Minimum Operable Instruments, place the inoperable instrument(s) in the tripped condition within 1 hour or declare the associated emergency diesel generator inoperable ~~and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as appropriate.~~

Action A

Action B

A.5

page 10 of 14

A.1

ITS 3.3.5.1

TABLE 3.3.3-1 (Continued)

M.2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
ACTION

add proposed Required Action D.1

ACTION 38 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

L.4

ACTION D

a. With one channel inoperable, remove the inoperable channel within 24 hours; restore the inoperable channel to OPERABLE status within 7 days or declare the associated ECCS systems inoperable.

b. With both channels inoperable, restore at least one channel to OPERABLE status within ~~one hour~~ or declare the associated ECCS system inoperable.

ACTION G

24 hours

LB.2

Table 3.3.5.1-1

TABLE 3.3.3-2

LF.1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTION

LA.1

LA.3

		TRIP FUNCTION	~~TRIP SETPOINT~~	ALLOWABLE VALUE
	A.	~~DIVISION 1 TRIP SYSTEM~~		
	1.	RHR-A (LPCI MODE) AND LPCS SYSTEM		
a.	a.	Reactor Vessel Water Level - Low Low Low, Level 1	~~≥ 129 inches*~~	≥ 136 inches*
b.	b.	Drywell Pressure - High	~~≤ 1.69 psig~~	≤ 1.89 psig
e	c.	LPCS Pump Discharge Flow-Low	~~≥ 750 gpm~~	≥ 640 gpm
g	d.	LPCS and LPCI A Injection Valve Injection Line-Low Pressure Interlock	~~500 psig~~	500 ± 20 psig
d	e.	LPCS and LPCI A Injection Valve Reactor Pressure-Low Pressure Interlock	~~500 psig~~	500 ± 20 psig
c.	f.	LPCI Pump A Start Time Delay Relay	~~≤ 5 seconds~~	≤ 6 seconds
f.	g.	LPCI Pump A Discharge Flow-Low	~~≥ 1000 gpm~~	≥ 550 gpm
h	h.	Manual Initiation	~~N.A.~~	N.A.
	2.	AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a.	a.	Reactor Vessel Water Level - Low Low Low, Level 1	~~≥ 129 inches*~~	≥ 136 inches*
b	b.	Drywell Pressure - High	~~≤ 1.69 psig~~	≤ 1.89 psig
c.	c.	Initiation Timer	~~≤ 105 seconds~~	≤ 117 seconds
d.	d.	Reactor Vessel Water Level-Low, Level 3	~~≥ 12.5 inches*~~	≥ 11 inches*
e.	e.	LPCS Pump Discharge Pressure-High	~~≥ 146 psig, increasing~~	≥ 136 psig, ~~increasing~~
f.	f.	LPCI Pump A Discharge Pressure-High	~~≥ 119 psig, increasing~~	≥ 106 psig, ~~increasing~~
h,	g.	Manual Initiation	~~N.A.~~	N.A.
g.	h.	Drywell Pressure Bypass Timer	~~< 9.0 minutes~~	~~Footnote (a)~~
	~~i.~~	~~Manual Inhibit~~	~~N.A.~~	~~N.A.~~

M.4 add upper limit

M.4 add lower limit

M.4 add upper limit

A.1

M.4 add upper limit

R.1

~~(a) The sum of the time delays associated with the ADS initiation timer and the drywell pressure bypass timer shall be less than or equal to 687 seconds.~~

M.5 add explicit limit

ITS 3.3.5.1

Table 3.3.5.1-1

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

LA.1 LF.1 A.1

		Function ~~TRIP~~ FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
	~~B.~~	~~DIVISION 2 TRIP SYSTEM~~ LA.3		
2.	1.	**RHR B AND C (LPCI MODE)**		
a.	a.	Reactor Vessel Water Level - Low Low Low, Level 1	~~>- 129 inches*~~	>- 136 inches*
b.	b.	Drywell Pressure - High	~~< 1.69 psig~~	< 1.89 psig
f.	c.	LPCI B and C Injection Valve Injection Line Low Pressure Interlock	~~500 psig~~	500 ± 20 psig
c.	d.	LPCI Pump B Start Time Delay Relay	~~< 5 seconds~~	< 6 seconds
e.	e.	LPCI Pump Discharge Flow-Low	~~> 1000 gpm~~	> 550 gpm
g.	f.	Manual Initiation	~~N.A.~~	N.A.
d.	g.	LPCI B and C Injection Valve Reactor Pressure-Low Pressure Interlock	~~500 psig~~	500 ± 20 psig
5.	2.	**AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"**		
a.	a.	Reactor Vessel Water Level - Low Low Low, Level 1	~~>- 129 inches*~~	>- 136 inches*
b.	b.	Drywell Pressure - High	~~< 1.69 psig~~	< 1.89 psig
c.	c.	Initiation Timer	~~< 105 seconds~~	< 117 seconds
d.	d.	Reactor Vessel Water Level-Low, Level 3	~~> 12.5 inches*~~	> 11 inches*
e.	e.	LPCI Pump B and C Discharge Pressure-High	~~> 119 psig, increasing~~	> 106 psig, ~~increasing~~
g.	f.	Manual Initiation	~~N.A.~~	N.A.
f.	g.	Drywell Pressure Bypass Timer	~~< 9.0 minutes~~	~~Footnote (a)~~
	~~h.~~	~~Manual Inhibit~~	~~N.A.~~	~~N.A.~~

M.4 add lower limit

M.4 add upper limit

add upper limit

add explicit limit

R.1

~~(a) The sum of the time delays associated with the ADS initiation timer and the drywell pressure bypass timer shall be less than or equal to 687 seconds.~~ M.5

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ITS 3.3.5.1

Table 3.3.5.1

TABLE 3.3.3-2 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

LF.1

LA.3

LA.1

Function TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
3 1. HPCS SYSTEM		
a. a. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
b. b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. c. Reactor Vessel Water Level - High, Level 8	≤ 55.5 inches*	≤ 56 inches*
~~d. Deleted~~		
~~e. Deleted~~		
d. f. Pump Discharge Pressure - High	≥ 120 psig	≥ 110 psig
e. g. HPCS System Flow Rate - Low	≥ 1000 gpm	≥ 900 gpm
f. h. Manual Intiation	N.A.	N.A.
D. LOSS OF POWER		
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)#		
a. 4.16 kV Buses		
1) Divisions 1 and 2	2625 ± 131 volts with ≤ 10 second time delay	2625 ± 262 volts with ≤ 11 second time delay
	2496 ± 125 volts with ≥ 4 second time delay	2496 ± 250 volts with ≥ 3 second time delay
2) Division 3	2870 ± 143 volts with ≤ 10 second time delay	2870 ± 287 volts with ≤ 11 second time delay

add upper limit

M.4

A.6 moved to ITS 3.3.8.1

A.1

TABLE NOTATIONS

A.10

*See Bases Figure B 3/4 3-1

#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

N.A. Not Applicable

A.6 moved to ITS 3.3.8.1

ITS 3.3.5.1

LOP A.2

Table 3.3.8.1-1

TABLE 3.3.3-2 (Continued)

~~EMERGENCY CORE COOLING SYSTEM ACTUATION~~ INSTRUMENTATION SETPOINTS

A.1

FUNCTION ~~TRIP FUNCTION~~	~~TRIP SETPOINT~~ LA.2	ALLOWABLE VALUE LF.1
C. DIVISION 3 TRIP SYSTEM <SEE ITS 3.3.5.1>		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	>- 50 inches*	>- 57 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 55.5 inches*	≤ 56 inches*
d. Deleted		
e. Deleted		
f. Pump Discharge Pressure - High	≥ 120 psig	≥ 110 psig
g. HPCS System Flow Rate - Low	≥ 1000 gpm	≥ 900 gpm
h. Manual Intiation	N.A.	N.A.
D. LOSS OF POWER		
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)#		
a. 4.16 kV Buses	LA.2	LF.1
1.a 1) Divisions 1 and 2	~~2625 ± 131 volts with ≤ 10 second time delay~~ ~~2496 ± 125 volts with ≥ 4 second time delay~~	2625 ± 262 volts with ≤ 11 second time delay 2496 ± 250 volts with ≥ 3 second time delay
2.a and 2.b 2) Division 3	~~2870 ± 143 volts with ≤ 10 second time delay~~	2870 ± 287 volts with ≥ 2 seconds time delay and ≤ 11 second time delay M.3

TABLE NOTATIONS

<SEE ITS 3.3.5.1> *See Bases Figure B 3/4 3-1.

LA.3 ~~#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.~~

N.A. Not Applicable <SEE ITS 3.3.5.1>

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER (Continued)		
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)		
a. 4.16 kV Buses		
1) Divisions 1, 2 and 3	≥3863 and ≤3877 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal	≥3814 and ≤3900 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal

A6

<Moved to ITS 3.3.8.1>

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ITS 3.3.5.1

LOP A.2

Table 3.3.8.1-1

TABLE 3.3.3-2 (Continued)

~~EMERGENCY CORE COOLING SYSTEM ACTUATION~~ INSTRUMENTATION SETPOINTS

Function TRIP FUNCTION	~~TRIP SETPOINT~~	ALLOWABLE VALUE
D. LOSS OF POWER (Continued)		LF.1
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)		
a. 4.16 kV Buses		
1) Divisions 1, 2 and 3 1.b, 1.c, 1.d, 2.c, 2.d, 2.e	~~≥3863 and ≤3877 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal~~	≥3814 and ≤3900 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal

LA.2

A.1

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ITS 3.3.8.1

A.1

ITS 3.3.5.1

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 60*, #
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	≤ 60*, #
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 41#
5. LOSS OF POWER	NA

A.2 moved to ITS 3.5.1 and 3.5.2

---

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

<SEE ITS 3.3.5.1>

A.1

ITS 3.3.8.1

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 60*,#
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	≤ 60*,#
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 41#
~~5. LOSS OF POWER~~	~~NA~~

A.3

<SEE ITS 3.3.5.1>

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

A.1

ITS 3.5.1

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	$\leq 60^{*,\#}$
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	$\leq 60^{*,\#}$
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	$\leq 41^{\#}$
5. LOSS OF POWER	NA

LA.4

---

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

SR 3.5.1.9 Note

ITS 3.5.2

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 60*,#
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	≤ 60*,#
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 41#
5. LOSS OF POWER	NA

LA.4

---

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

SR 3.5.2.7 NOTE

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Table 3.3.5.1-1

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTION TRIP FUNCTION	SR 3.3.5.1.1 CHANNEL CHECK	SR 3.3.5.1.2 CHANNEL FUNCTIONAL TEST	SR 3.3.5.1.3 SR 3.3.5.1.4 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM LA.3			24 months LE.1	
1 1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	4-R	1, 2, 3, 4*, 5*
b. b. Drywell Pressure - High	NA	Q	4-Q	1, 2, 3
e. c. LPCS Pump Discharge Flow-Low	NA	Q	Q-3	1, 2, 3, 4*, 5*
g. d. LPCS and LPCI A Injection Valve Injection Line Pressure Low Interlock	NA	Q	4-R	1, 2, 3, 4*, 5*
d. e. LPCS and LCPI A Injection Valve Reactor Pressure Low Interlock	NA	Q	4-R	1, 2, 3, 4*, 5*
c. f. LPCI Pump A Start Time Delay Relay	NA	Q	Q-3	1, 2, 3, 4*, 5*
f. g. LPCI Pump A Flow-Low	NA	Q	Q-3	1, 2, 3, 4*, 5*
h. h. Manual Initiation	NA	LD.1 R A.9	NA	1, 2, 3, 4*, 5*
4 2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				A.1
a. a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	4-R	1, 2, 3
b. b. Drywell Pressure-High	NA	Q	4-Q	1, 2, 3
g. c. Initiation Timer	NA	Q	Q-3	1, 2, 3
d. d. Reactor Vessel Water Level - Low, Level 3	S	Q	4-R	1, 2, 3
e. e. LPCS Pump Discharge Pressure-High	NA	Q	4-Q	1, 2, 3
f. f. LPCI Pump A Discharge Pressure-High	NA	Q	4-Q	1, 2, 3
h. g. Manual Initiation	NA	LD.1 R A.9	NA	1, 2, 3
g h. Drywell Pressure Bypass Timer	NA	Q	Q-3	1, 2, 3
i. Manual Inhibit	NA	R	NA	1, 2, 3 R.1

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ITS 3.3.5.1

Table 3.3.5.1-1

TABLE ...3.1-1 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Function TRIP FUNCTION	SR 3.3.5.1.1 CHANNEL CHECK	SR 3.3.5.1.2 CHANNEL FUNCTIONAL TEST	SR 3.3.5.1.3 SR 3.3.5.1.4 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
	B. DIVISION 2 TRIP SYSTEM LA.3			24 months LE.1	
2	1. RHR B AND C (LPCI MODE)				
a.	a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R-4	1, 2, 3, 4*, 5*
b.	b. Drywell Pressure - High	NA	Q	Q-4	1, 2, 3
f.	c. LPCI B and C Injection Valve Injection Line Pressure Low Interlock	NA	Q	R-4	1, 2, 3, 4*, 5*
c.	d. LPCI Pump B Start Time Delay Relay	NA	Q	Q-3	1, 2, 3, 4*, 5*
e.	e. LPCI Pump Discharge Flow-Low	NA	Q	Q-3	1, 2, 3, 4*, 5*
g.	f. Manual Initiation	NA	LD.1 R A.9	NA	1, 2, 3, 4*, 5*
d	g. LPCI B and C Injection Valve Reactor Pressure Low Interlock	NA	Q	R-4	1, 2, 3, 4*, 5*
5	2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a.	a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R-4	1, 2, 3
b	b. Drywell Pressure-High	NA	Q	Q-4	1, 2, 3
c.	c. Initiation Timer	NA	Q	Q-3	1, 2, 3
d.	d. Reactor Vessel Water Level - Low, Level 3	S	Q	R-4	1, 2, 3
e.	e. LPCI Pump B and C Discharge Pressure-High	NA	Q	Q-4	1, 2, 3
g.	f. Manual Initiation	NA	LD.1 R A.9	NA	1, 2, 3
f.	g. Drywell Pressure Bypass Timer	NA	Q	Q-3	1, 2, 3
	h. Manual Inhibit	NA	R	NA	1, 2, 3 R.1

A.1

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ITS 3.3.5.1

Table 3.3.5.1-1

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	SR 3.3.5.1.1 CHANNEL CHECK	SR 3.3.5.1.2 CHANNEL FUNCTIONAL TEST	SR 3.3.5.1.3 SR 3.3.5.1.4 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
~~C. DIVISION 3/TRIP SYSTEM~~ LA.3				
3, 1. HPCS SYSTEM			24 months LE.1	
a, a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	~~R~~ -4	1, 2, 3, 4*, 5*
b, b. Drywell Pressure-High	NA	Q	~~Q~~-4	1, 2, 3
c, c. Reactor Vessel Water Level-High Level 8	S	Q	~~R~~ -4	1, 2, 3, 4*, 5*
~~d. Deleted~~				
~~e. Deleted~~				
d. f. Pump Discharge Pressure-High	NA	Q	~~Q~~-4	1, 2, 3, 4*, 5*
e, g. HPCS System Flow Rate-Low	NA	Q	Q -3	1, 2, 3, 4*, 5*
f, h. Manual Initiation	NA	LD.1 ~~R~~ A.7	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

A.6 moved to ITS 3.3.8.1

TABLE NOTATIONS

Note (c) to Table 3.3.5.1-1

#Not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~122~~ 150 psig. L.2

Note (a) to Table 3.3.5.1-1

*When the system is required to be OPERABLE after being manually realigned, as applicable, per Specification 3.5.2.

**Required when ESF equipment is required to be OPERABLE. A.6 moved to ITS 3.3.8.1

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ITS 3.3.5.1

LOP A.2

Table 3.3.8.1-1

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

SR 3.3.8.1.1
SR 3.3.8.1.2

FUNCTION TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
<SEE ITS 3.3.5.1>				
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	Q	Q	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	S	Q	R	1, 2, 3, 4*, 5*
d. Deleted				
e. Deleted				
f. Pump Discharge Pressure-High	NA	Q	Q	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	NA	Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1.a, 2.a, 2.b 1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
1.b, 1.c, 1.d, 2.c, 2.d, 2.e 2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

24 months

LD.1 LE.1

M.1

A.1

<SEE ITS 3.3.5.1>

TABLE NOTATIONS

*Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 122 psig.
*When the system is required to be OPERABLE after being manually realigned, as applicable, per Specification 3.5.2.

**Required when ESF equipment is required to be OPERABLE.

Applicability

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ITS 3.3.8.1

A.1

Specification 3.3.4.2

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.4.2

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.1-2. LA.1

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: Add proposed ACTIONS Note A.2

ACTIONS A, B, and C

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value. LA.1

ACTION A

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 24 hours. Add proposed Required Action A.1 A.3 14 days Add proposed Note to Required Action A.2

Condition A

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and: LB.1 Add proposed Note to Required Action A.2 M.1

Required Action A.2

1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within 24 hours, or, if this action will initiate a pump trip, declare the trip system inoperable. 14 days LB.1

Required Action A.1

2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable. L.1

ACTION B d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within ACTION D the next 6 hours. function L.2 Add proposed Required Action D.1 L.3

ACTION C e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the ACTION D next 6 hours. function L.2 Add proposed Required Action D.1 L.3

SURVEILLANCE REQUIREMENTS

SRs 3.3.4.2.1, 3.3.4.2.2, 3.3.4.2.3

4.3.4.1.1 Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

SR 3.3.4.2.4

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months. A.4 24 LD.1

A1

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

	TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM [(a)]
LCO 3.3.4.2.a	1. Reactor Vessel Water Level - Low Low, Level 2	2
LCO 3.3.4.2.b	2. Reactor Vessel Pressure-High	2

Note to Surveillance Requirements

(a) One channel in one trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that all other channels are OPERABLE.

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Specification 3.3.4.2

A.1

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

	TRIP FUNCTION	~~TRIP SETPOINT~~ LA.1	ALLOWABLE VALUE LF.1
LCO 3.3.4.2.a	1. Reactor Vessel, Water Level - Low Low, Level 2	~~≥ -50 inches*~~ A.5	SR 3.3.4.2.3 ≥ -57 inches* A.5
LCO 3.3.4.2.b	2. Reactor Vessel Pressure-High	~~≤ 1135 psig~~	SR 3.3.4.2.3 ≤ 1150 psig

---

~~* See Bases Figure B3/4 3-1.~~ A.5

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Specification 3.3.4.2

A.1

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

		SR 3.3.4.2.1	SR 3.3.4.2.2	SR 3.3.4.2.3
	TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
LCO 3.3.4.2.a	1. Reactor Vessel Water Level - Low Low, Level 2	S	Q	~~R~~ 24 months LE.1
LCO 3.3.4.2.b	2. Reactor Vessel Pressure - High	S	Q	~~Q~~ 24 months

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Specification 3.3.4.2

A.1 Specification 3.3.4.1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.3.4.1.b A.2

LCO 3.3.4.1.a

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3. LA.3 LA.2

SR 3.3.4.1.5

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER with any recirculation pump in fast speed. A.3 L.2

ACTION: add proposed ACTIONS Note A.4

ACTIONS A and B

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value. LA.3

ACTION A

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 12 72 hours. LB.1

CONDITION A

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement(s) for one trip system and: M.1

add proposed Required Action A.1 and Required Action A.2 Note

Required Action A.2

1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 72 hours. LB.1

Required Action A.1

2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable. L.3

ACTIONS A and B

d. With one trip system function inoperable, restore the inoperable trip system function to OPERABLE status within 72 hours, otherwise, either: L.4

Required Action B.2

1. Increase the MINIMUM CRITICAL POWER (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 2 hour, or L.1

Required Action C.2

2. Reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 6 4 hours. M.2

ACTIONS A and B

e. With both trip systems function inoperable, restore at least one trip system to OPERABLE status within 1 2 hours otherwise, either: L.4 L.1

Required Action B.2

1. Increase the MINIMUM CRITICAL POWER (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 2 hour, or L.1

Required Action C.2

2. Reduce THERMAL POWER to less than 25% RATED THERMAL POWER within the next 6 4 hours. M.2

Add Required Action C.1 L.2

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INSTRUMENTATION

A.1

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.1 SR 3.3.4.1.2 4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

A.5

SR 3.3.4.1.3 4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months. 24 LD.1

SR 3.3.4.1.5 4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function ~~shown in Table 3.3.4.2-3~~ shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. SR 3.3.4.1.6 The time allotted for breaker arc suppression shall be verified by test at least once per 60 months.

LA.3

LA.4

48 LD.1

Add Note 1 to SR 3.3.4.1.5 A.6

Add Note 2 to SR 3.3.4.1.5 A.7

A.1

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LCO 3.3.4.1.a

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEMⓐ
1. Turbine Stop Valve Closure	2ⓑ
2. Turbine Control Valve - Fast Closure	2ⓑ

Note to Surveillance Requirement

LA.1

Note to Surveillance Requirements

(a) A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

(b) This function shall not be automatically bypassed when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

LA.1

Specification 3.3.4.1

A.1

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

LCO 3.3.4.1.a

SR 3.3.4.1.2

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	~~≤ 5% closed~~	≤ 7% closed
2. Turbine Control Valve - Fast Closure	~~≥ 500 psig~~	≥ 414 psig

LA3

LF.1

Specification 3.3.4.1

Specification 3.3.4.1

A.1

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milliseconds)
1. Turbine Stop Valve-Closure	≤ 97
2. Turbine Control Valve - Fast Closure	≤ 97

LA.2

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A.1

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

LCO 3.3.4.1.a TRIP FUNCTION	SR 3.3.4.1.1 CHANNEL FUNCTIONAL TEST	SR 3.3.4.1.2 CHANNEL CALIBRATION
1. Turbine Stop Valve Closure(a)	Q	R
2. Turbine Control Valve-Fast Closure(a)	Q	R

24 months LE.1

24 LD.1

SR 3.3.4.1.4 (a) At least once per ~~18~~ 24 months, verify Turbine Stop Valve - Closure and Turbine Control Valve - Fast Closure Trip Functions are not bypassed when THERMAL POWER is ≥ 25% of RATED THERMAL POWER.

SR 3.0.2 Specification 4.0.2 applies to this ~~18 month~~ interval. LD.1

ITS 3.3.5.2

A.1

INSTRUMENTATION

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.5.2 — 3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2. — LA.3

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION: add proposed Actions Note — A.2

ACTION A — a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value. — LA.3

ACTION A — b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

Note 1 to Surveillance Requirements — 4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

SR 3.3.5.2.4 — 4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months. — LA.1

24 — LD.1

A.1

Table 3.3.5.2-1

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

A.3

Note 2 to Surveillance Requirements

FUNCTION FUNCTIONAL UNITS	FUNCTION MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
1. a. Reactor Vessel Water Level - Low Low, Level 2	~~2~~ 4	50 B, E
2. b. Reactor Vessel Water Level - High, Level 8	2 ~~(b)~~	51 C, E
4. c. Manual Initiation	1 ~~(c)~~	52 C, E

A.3

LA.2

add proposed Function 3

M.1

for Functions 1 and 3 only

LB.1

Note 2 to Surveillance Requirements

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

~~(b) One trip system with two-out-of-two logic.~~

~~(c) Single channel.~~

LA.2

ITS 3.3.5.2

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ITS 3.3.5.2

TABLE 3.3.5-1 (Continued) A.1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

add proposed Required Action B.1 — M.2

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

Action B

a. For one trip system, place the inoperable channel in the tripped condition within 24 hours or declare the RCIC system inoperable.

Action E

Actions B and E

b. For both trip systems, declare the RCIC system inoperable. — L.1

ACTION 51 - With the number of OPERABLE channels less than required by the minimum OPERABLE Channels per Trip System requirement, declare the RCIC system inoperable within 24 hours.

Action C

Action E

restore channel — A4

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the RCIC system inoperable.

Action C

Action E

add proposed ACTION D — M.1

A.1

Table 3.3.5.2-1

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTION FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE	
1. a. Reactor Vessel Water Level - Low Low, Level 2	~~≥ - 50 inches*~~	≥ - 57 inches*	LF.1
2. b. Reactor Vessel Water Level - High, Level 8	~~≤ 55.5 inches*~~	≤ 56 inches*	
4. c. Manual Initiation	~~NA~~	NA	

LA.3

add proposed function 3 — M.2

~~*See Bases Figure B 3/4 3-1~~ — A.6

ITS 3.3.5.2

A.1

Table 3.3.5.2-1
TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTION FUNCTIONAL UNITS	SR 3.3.5.2.1 CHANNEL CHECK	SR 3.3.5.2.2 CHANNEL FUNCTIONAL TEST	SR 3.3.5.2.3 CHANNEL CALIBRATION
1. a. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	~~R~~ 24 months
2. b. Reactor Vessel Water Level - High, Level 8	S	Q	~~R~~ 24 months
4. c. Manual Initiation	NA	~~R~~	NA

LE.1

A.5

add proposed Function 3

M.2

ITS 3.3.5.2

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3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION A.1

ITS 3.3.2.1

LIMITING CONDITION FOR OPERATION

LCO 3.3.2.1 3.3.6 The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.~~ LA.3

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

ACTIONS A AND B a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status ~~with its trip setpoint adjusted consistent with the Trip Setpoint value.~~ LA.3

ACTIONS A AND B b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

Note 1 to Surveillances 4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST* and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

M.1

Note 2 to Surveillances * A channel may be placed in an inoperable status for up to 6 hours for required surveillance ~~or 12 hours for repair~~ without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

Table 3.3.2.1-1

TABLE 3.3.6-1
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LA.1

A.2

	Function TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
	1. ROD BLOCK MONITOR~~(a)~~			
1.a	a. Upscale	2	1*	60
1.b	b. Inoperative	2	1*	60
1.c	c. Downscale	2	1*	60
	~~2. APRM~~			
	~~a. Flow Biased Simulated Thermal Power-Upscale~~	~~4~~	~~1~~	~~61~~
	~~b. Inoperative~~	~~4~~	~~1, 2, 5~~	~~61~~
	~~c. Downscale~~	~~4~~	~~1~~	~~61~~
	~~d. Neutron Flux-High~~	~~4~~	~~2, 5~~	~~61~~
	~~3. SOURCE RANGE MONITORS~~			
	~~a. Detector not full in(b)~~	~~3~~ ~~2~~	~~2~~ ~~5~~	~~61~~ ~~61~~
	~~b. Upscale(c)~~	~~3~~ ~~2~~	~~2~~ ~~5~~	~~61~~ ~~61~~
	~~c. Inoperative(c)~~	~~3~~ ~~2~~	~~2~~ ~~5~~	~~61~~ ~~61~~
	~~d. Downscale(d)~~	~~3~~ ~~2~~	~~2~~ ~~5~~	~~61~~ ~~61~~
	~~4. INTERMEDIATE RANGE MONITORS~~			
	~~a. Detector not full in~~	~~6~~	~~2, 5~~	~~61~~
	~~b. Upscale~~	~~6~~	~~2, 5~~	~~61~~
	~~c. Inoperative~~	~~6~~	~~2, 5~~	~~61~~
	~~d. Downscale(e)~~	~~6~~	~~2, 5~~	~~61~~
	~~5. SCRAM DISCHARGE VOLUME~~			
	~~a. Water Level-High~~	~~2~~	~~1, 2, 5**~~	~~62~~
	~~b. Scram Discharge Volume Switch in Bypass~~	~~1~~	~~5**~~	~~62~~
	~~6. RECIRCULATION FLOW UNIT~~			
	~~a. Upscale~~	~~2~~	~~1~~	~~62~~
	~~b. Inoperative~~	~~2~~	~~1~~	~~62~~
	~~c. Comparator~~	~~2~~	~~1~~	~~62~~

ACTIONS A and B

R.1

A.1

M.5

Add proposed Function 3 of Table 3.3.2.1-1

1

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ITS 3.3.2.1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTIONS A and B — ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3. R.1

~~ACTION 61 - With the number of OPERABLE channels:~~

~~a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.~~

~~b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.~~

~~ACTION 62 - With the number of OPERABLE Channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 12 hours.~~

NOTE

Table 3.3.2.1-1 Note (a) * With THERMAL POWER ≥ 30% of RATED THERMAL POWER and no peripheral Control rod selected. A.2

~~** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ R.1

~~a. The RBM shall be automatically bypassed when a peripheral control rod is selected.~~ LA.1

~~b. This function shall be automatically bypassed if detector count rate is ≥ 100 cps or the IRM channels are on range 3 or higher.~~ R.1

~~c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.~~

~~d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.~~

~~e. This function shall be automatically bypassed when the IRM channels are on range 1.~~

Add proposed ACTION E — M.5

Table 3.3.2.1-1

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

LF.1

A.1

	Function TRIP FUNCTION	TRIP SETPOINT LA.3	ALLOWABLE VALUE
	1. **ROD BLOCK MONITOR**		
1.a	a. Upscale The Rod Block Monitor Upscale Setpoints shall be established according to the relationships specified in the CORE OPERATING LIMITS REPORT.		
1.b	b. Inoperative	N.A.	N.A.
1.c	c. Downscale	≥ 5% of RATED THERMAL POWER LA.3	≥ 3% of RATED THERMAL POWER LF.1
R.1	~~2. **APRM**~~		
	~~a. Flow Biased Simulated Thermal Power-Upscale~~		
	~~1) Two Recirculation Loop Operation~~	~~≤ 0.58 W + 47%*~~	~~≤ 0.58 W + 50%*~~
	~~2) Single Recirculation Loop Operation~~	~~< 0.58 W + 42.3%~~	~~< 0.58 W + 45.3%*~~
	~~b. Inoperative~~	~~N.A.~~	~~N.A.~~
	~~c. Downscale~~	~~≥ 5% of RATED THERMAL POWER~~	~~≥ 3% of RATED THERMAL POWER~~
	~~d. Neutron Flux-High~~	~~≤ 12% of RATED THERMAL POWER~~	~~≤ 14% of RATED THERMAL POWER~~
	~~3. **SOURCE RANGE MONITORS**~~		
	~~a. Detector not full in~~	~~N.A.~~	~~N.A.~~
	~~b. Upscale~~	~~$< 2 \times 10^5$ cps~~	~~$< 5 \times 10^5$ cps~~
	~~c. Inoperative~~	~~N.A.~~	~~N.A.~~
	~~d. Downscale~~	~~≥ 0.7 cps~~	~~≥ 0.5 cps~~
	~~4. **INTERMEDIATE RANGE MONITORS**~~		
	~~a. Detector not full in~~	~~N.A.~~	~~N.A.~~
	~~b. Upscale~~	~~< 108/125 of full scale~~	~~< 110/125 of full scale~~
	~~c. Inoperative~~	~~N.A.~~	~~N.A.~~
	~~d. Downscale~~	~~≥ 5/125 of full scale~~	~~≥ 3/125 of full scale~~

TABLE 3.3.6-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

R.1

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 765' 5¼"	≤ 765' 5¼"
b. Scram Discharge Volume Switch in Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 of full scale	≤ 111/125 of full scale
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

---

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W).

A.1

ITS 3.3.2.1

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Table 3.3.2.1-1

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

L.1 LA.1 A.2 R.1

Function TRIP FUNCTION	CHANNEL CHECK	SR 3.3.2.1.1 CHANNEL FUNCTIONAL TEST	SR 3.3.2.1.4 CHANNEL CALIBRATION(a) Note to SR 3.3.2.1.4	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
1.a a. Upscale	N.A.	~~S/U$^{(b)(c)}$~~,Q~~$^{(c)}$~~	Q	1*
1.b b. Inoperative	N.A.	~~S/U$^{(b)(c)}$~~,Q~~$^{(c)}$~~	N.A.	1*
1.c c. Downscale	N.A.	~~S/U$^{(b)(c)}$~~,Q~~$^{(c)}$~~	Q	1*
~~2. APRM~~				
~~a. Flow Biased Simulated Thermal Power-Upscale~~	~~N.A.~~	~~S/U$^{(b)}$,Q~~	~~SA~~	~~1~~
~~b. Inoperative~~	~~N.A.~~	~~S/U$^{(b)}$,Q~~	~~N.A.~~	~~1, 2, 5~~
~~c. Downscale~~	~~N.A.~~	~~S/U$^{(b)}$,Q~~	~~SA~~	~~1~~
~~d. Neutron Flux-High~~	~~N.A.~~	~~S/U$^{(b)}$,Q~~	~~SA~~	~~2***, 5~~
~~3. SOURCE RANGE MONITORS~~				
~~a. Detector not full in~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~N.A.~~	~~2***, 5~~
~~b. Upscale~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~Q~~	~~2***, 5~~
~~c. Inoperative~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~N.A.~~	~~2***, 5~~
~~d. Downscale~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~Q~~	~~2***, 5~~
~~4. INTERMEDIATE RANGE MONITORS~~				
~~a. Detector not full in~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~N.A.~~	~~2***, 5~~
~~b. Upscale~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~Q~~	~~2***, 5~~
~~c. Inoperative~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~N.A.~~	~~2***, 5~~
~~d. Downscale~~	~~N.A.~~	~~S/U$^{(b)}$,W~~	~~Q~~	~~2***, 5~~
~~5. SCRAM DISCHARGE VOLUME~~				
~~a. Water Level-High~~	~~N.A.~~	~~Q~~	~~R~~	~~1, 2, 5**~~
~~b. Scram Discharge Volume Switch in Bypass~~	~~N.A.~~	~~Q~~	~~N.A.~~	~~5**~~
~~6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW~~				
~~a. Upscale~~	~~N.A.~~	~~S/U$^{(b)}$,Q~~	~~Q~~	~~1~~
~~b. Inoperative~~	~~N.A.~~	~~S/U$^{(b)}$,Q~~	~~N.A.~~	~~1~~
~~c. Comparator~~	~~N.A.~~	~~S/U$^{(b)}$,Q~~	~~Q~~	~~1~~

A.1

Add proposed SR 3.3.1.2.5 M.2

Add proposed SR 3.3.1.2.7 M.5

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ITS 3.3.2.1

TABLE 4.3.6-1 (Continued) A.1 ITS 3.3.2.1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

Note to SR3.3.2.1.4 (a) Neutron detectors may be excluded from CHANNEL CALIBRATION. L.1 for RBM

~~(b) Within 24 hours prior to startup, if not performed within the previous 7 days.~~ R.1 for Functions 2-6

~~(c) Includes reactor manual control multiplexing system input.~~ LA.2

Table 3.3.2.1-1 Note(a) * With THERMAL POWER ≥ 30% of RATED THERMAL POWER. and no peripheral Control rod selected A.2

~~** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~

~~*** The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.~~

R.1

ITS 3.3.7.1

INSTRUMENTATION A.1

3/4.3.7 MONITORING INSTRUMENTATION

~~RADIATION MONITORING~~ INSTRUMENTATION

Control Room Area Filtration (CRAF) A.2

CRAF System

LIMITING CONDITION FOR OPERATION

LCO 3.3.7.1 3.3.7.1 The ~~radiation monitoring~~ instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE* ~~with their alarm/trip setpoints within the specified limits~~. A.2

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION: add proposed ACTIONS NOTE A.3

ACTION A a. With a radiation monitoring instrumentation channel ~~alarm/trip setpoint~~ exceeding the value shown in Table 3.3.7.1-1, ~~adjust the setpoint to within the limit within 4 hours or~~ declare the channel inoperable. M.1

ACTION A b. With one or more ~~radiation monitoring~~ CRAF System channels inoperable, take the ACTION required by Table 3.3.7.1-1. A.2

~~c. The provisions of Specification 3.0.3 are not applicable.~~ A.4

CRAF System A.2

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required ~~radiation monitoring~~ instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

SR 3.3.7.1.1
SR 3.3.7.1.2
SR 3.3.7.1.3

A.5

~~*The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4 or 5 or when defueled.~~

A.2

Allowable Value

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

SR 3.3.7.1.3

LA.1

L(O 3.3.7.1)

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
a. Main Control Room Atmospheric Control System Radiation Monitoring Subsystem	2 per trip system/train (intake)**	1,2,3,5 and *	3.5 mR/hr	0.1 to 10,000 mR/hr	70 A, B

L.1 LF.1

add proposed 3rd and 4th Applicability

A.1

TABLE NOTATIONS

2nd Applicability

*When irradiated fuel is being handled in the secondary containment.

Note to Surveillance Requirements

**A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition, ~~provided at least one other operable channel in the same Trip System is monitoring that Trip Function~~.

M.2

place in trip in 6 hours L.2

ACTION STATEMENT

add proposed Required Action A.1

ACTION A

ACTION 70 -

a. With the number of OPERABLE channels per trip system one less than the minimum required, place the inoperable channel in the tripped condition within ~~one~~ 6 ~~hour~~. 5 LB.1

ACTION B

b. With both channels in a trip system inoperable, ~~declare the trip system inoperable~~. ~~Restore the inoperable trip system to OPERABLE status within 7 days,~~ or, within the next ~~6~~ 1 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation.

L.2

L.2

c. Otherwise, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within 1 hour.

add proposed Required Action B.2

L.2

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ITS 3.3.7.1

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

		INSTRUMENTATION	SR 3.3.7.1.1 CHANNEL CHECK	SR 3.3.7.1.2 CHANNEL FUNCTIONAL TEST	3.3.7.1.3 CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
LCO 3.3.7.1	a.	Main Control Room Atmospheric Control System Radiation Monitoring Subsystem	S	Q	~~R~~ 24 months LE.1	L.1 1,2,3, ~~5~~ and * add proposed 3rd and 4th Applicability L.1

NOTES

2nd Applicability *When irradiated fuel is being handled in the secondary containment.

A.1

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ITS 3.3.7.1

METEOROLOGICAL MONITORING INSTRUMENTATION*

LIMITING CONDITION FOR OPERATION

3.3.7.3 The meteorological monitoring instrumentation channels shown in Table 3.3.7.3-1 shall be OPERABLE.**

APPLICABILITY: At all times.

ACTION:

a. With one or more meteorological monitoring instrumentation channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.3 Each of the above required meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.3-1.

---

*The Meteorological Monitoring Instrumentation System is shared between La Salle Unit 1 and La Salle Unit 2.

**The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4 or 5 or when defueled.

CTS 3/4.3.7.3

~~TABLE 3.3.7.3-1~~

~~METEOROLOGICAL MONITORING INSTRUMENTATION~~

~~INSTRUMENT~~	~~MINIMUM INSTRUMENTS OPERABLE~~
~~a. Wind Speed~~	
~~1. Elev. 200 ft. and 375 ft.~~	~~1 each~~
~~b. Wind Direction~~	
~~1. Elev. 200 ft. and 375 ft.~~	~~1 each~~
~~c. Air Temperature Difference~~	
~~1. Elev. 33/200 ft. or Elev. 33/375 ft~~	~~1~~

R.1

Page 5 of 6

CTS 3/4.3.7.3

R.1

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 200 ft. and 375 ft.	D	SA
b. Wind Direction		
1. Elev. 200 ft. and 375 ft.	D	SA
c. Air Temperature Difference		
1. Elev. 33/200 ft. or 33/375 ft.	D	SA

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A.1

INSTRUMENTATION

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.3.2 3.3.7.4 The remote shutdown monitoring instrumentation channels shown in Table 3.3.7.4-1 shall be OPERABLE with readouts displayed in the remote shutdown panel external to the control room. LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: Add proposed ACTIONS Note 2 A.2

LA.1

Action A a. With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable

Action B channel(s) to OPERABLE status within X days or be in at least HOT SHUTDOWN within the next 12 hours. 30 L.1

Note 1 to Actions b. The provisions of Specification 3.0.4 are not applicable.

L.2

Add Proposed Note to Surveillance Requirements

L.3

for each required instrumentation channel that is normally energized

SURVEILLANCE REQUIREMENTS

SR 3.3.3.2.1 4.3.7.4 Each of the above required remote shutdown monitoring instrumentation

SR 3.3.3.2.2 channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

LA.1

A.1

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
1. Reactor Vessel Pressure	1
2. Reactor Vessel Water Level	1
3. RHR Flow	1
4. RHR Service Water Flow	1
5. RHR Service Water Temperature	1
6. RCIC Flow	1
7. RCIC Turbine Speed	1
8. Suppression Pool Water Level	1
9. Suppression Pool Water Temperature	1

LA.1

LA.1

A.1

ITS 3.3.3.2

~~TABLE 4.3.7.4-1~~

~~REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

~~INSTRUMENT~~	SR 3.3.3.2.1 CHANNEL CHECK	SR 3.3.3.2.2 CHANNEL CALIBRATION
~~1. Reactor Vessel Pressure~~	M	R
~~2. Reactor Vessel Water Level~~	M	R
~~3. RHR Flow~~	M	R
~~4. RHR Service Water Flow~~	M	R
~~5. RHR Service Water Temperature~~	M	R
~~6. RCIC Flow~~	M	R
~~7. RCIC Turbine Speed~~	M	R
~~8. Suppression Pool Water Level~~	M	R
~~9. Suppression Pool Water Temperature~~	M	R

L.3

24 months

LE.1

LD.1

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INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.3.1 3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

Add proposed ACTIONS NOTE 1 L.1

Add proposed ACTIONS NOTE 2 A.2

ACTIONS A - F

a. With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

Note 1 to Surveillance Requirements

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

Add proposed Note 2 to Surveillance Requirements L.2

TABLE 3.3.3.1-1

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

A.5

	FUNCTION	REQUIRED NUMBER OF CHANNELS	~~MINIMUM CHANNELS OPERABLE~~	ACTION	
1.	1. Reactor Vessel Pressure	2	~~1~~	80 A,B,C,E	A.1
2.	2. Reactor Vessel Water Level a. Fuel Zone b. Wide Range M.3	②	~~1~~	80 A,B,C,E	
3.	3. Suppression Chamber Water Level	2	~~1~~	80 A,B,C,E	
9.	4. Suppression Chamber Water Temperature	M.2 ② ~~7, 1/well~~	~~7, 1/well~~	80 A,B,C,E	
	~~5. Suppression Chamber Air Temperature~~	~~2~~	~~1~~	~~80~~	R.1
4.	6. Drywell Pressure a. Narrow Range b. Wide Range M.4	②	~~1~~ A.5	80 A,B,C,E	
	~~7. Drywell Air Temperature~~	~~2~~	~~1~~	~~80~~	R.1
7.	8. Drywell Oxygen Concentration* LA.1	2	~~1~~ A.5	80 A,B,C,E	
8.	9. Drywell Hydrogen Concentration Analyzer* and Monitor	2	~~1~~	82 A,B,C,E	
5.	10. Primary Containment Gross Gamma Radiation	2	~~1~~	81 A,B,C,F	
	~~11. Safety/Relief Valve Position Indicators~~	~~1/valve~~	~~1/valve~~	~~80~~	R.1
	~~12. Noble Gas Monitor, Main Stack~~	~~1~~	~~1~~	~~81~~	
	~~13. Noble Gas Monitor, Standby Gas Treatment System Stack~~	~~1~~	~~1~~	~~81~~	

~~*Actuated after LOCA.~~ LA.1

Add proposed Function 6 M.1

A.1

ITS 3.3.3.1

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

ACTION A a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within ~~7~~ 30 days ~~or be in at least HOT SHUTDOWN within the next 12 hours.~~ L.4

L.3

add proposed ACTION B L.4

ACTION C b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within ~~48 hours~~ 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

ACTIONS D and E

L.3

ACTION 81 -

LA.2 ~~With the number of OPERABLE channels less than the required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours~~, and: with two required channels inoperable

ACTION C 1) ~~either~~ restore the ~~inoperable~~ one required channel~~(s)~~ to OPERABLE status within 7 days of the event, or

Insert proposed ACTION A L.5

ACTIONS B, D, and F 2) prepare and submit a Special Report to the Commission pursuant to Specification 6.6.c within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

A.3 moved to ITS 5.6

ACTION 82 -

ACTION A a. With the number of OPERABLE channels one less than the required number of channels shown in Table 3.3.7.5-1, restore the inoperable channel to OPERABLE status within 30 days ~~or be in at least HOT SHUTDOWN within the next 12 hours.~~ L.4

Add proposed ACTION B

ACTION C b. With the number of OPERABLE channels less then the minimum channels OPERABLE requirements of Table 3.3.7.5-1, restore at least one channel to OPERABLE status within 7 days ~~or be~~ in at least HOT SHUTDOWN within the next 12 hours.

ACTIONS D and E

SEE ITS 3.3.3.1

A.1

ITS 5.6

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 -

With the number of OPERABLE channels less than the required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

5.6.6 2) prepare and submit a Special Report to the Commission pursuant to Specification 6.6.c within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 -

a. With the number of OPERABLE channels one less than the required number of channels shown in Table 3.3.7.5-1, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less then the minimum channels OPERABLE requirements of Table 3.3.7.5-1, restore at least one channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SEE ITS 3.3.3.1

Table 3.3.3.1-1

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function	INSTRUMENT	SR 3.3.3.1.1 CHANNEL CHECK	SR 3.3.3.1.2, SR 3.3.3.1.3 CHANNEL CALIBRATION	
1.	1. Reactor Vessel Pressure	M	3-~~R~~	24 months LE.1 LD.1
2.	2. Reactor Vessel Water Level	M	3-~~R~~	24 months LE.1 LD.1
3.	3. Suppression Chamber Water Level	M	3-~~R~~	24 months LE.1 LD.1
9.	4. Suppression Chamber Water Temperature	M	3-~~R~~	24 months LE.1 LD.1
	~~5. Suppression Chamber Air Temperature~~	~~M~~	~~R~~	R.1
4.	6. Primary Containment Pressure	M	3-R	
	~~7. Drywell Air Temperature~~	~~M~~	~~R~~	R.1
7.	8. Drywell Oxygen Concentration	M	2-~~R~~ Q	M.5
8.	9. Drywell Hydrogen Concentration Analyzer and Monitor	M	2-Q	
5.	10. Primary Containment Gross Gamma Radiation	M	3-~~R~~	24 months LE.1 LD.1
	~~11. Safety/Relief Valve Position Indicators~~	~~M~~	~~R~~	R.1
	~~12. Noble Gas Monitor, Main Stack~~	~~M~~	~~R~~	R.1
	~~13. Noble Gas Monitor, Standby Gas Treatment System Stack~~	~~M~~	~~R~~	R.1

A.1

Add proposed FUNCTION 6 M.1

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ITS 3.3.3.1

A.1

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

two for MODES 3 and 4 — L.6

LCO 3.3.1.2 Table 3.3.1.2-1 { 3.3.7.6 At least three source range monitor channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3, and 4.

ACTION:

ACTION A { a. In OPERATIONAL CONDITION 2* with one (or more — L.1) of the above required source range monitor channels inoperable, restore at least three source range monitor channels to OPERABLE status within 4 hours or be in at least

ACTION C — HOT SHUTDOWN within the next 12 hours. ← Add proposed ACTION B — L.1

ACTION D { b. In OPERATIONAL CONDITION 3 or 4 with two (L.6) or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock (place — L.2) the reactor (A.2) mode switch in the Shutdown position within 1 hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by: ← Add proposed Note to Surveillance Requirements — A.3

a. Performance of a:

1. CHANNEL CHECK at least once per:

SR 3.3.1.2.1 a) 12 hours in CONDITION 2*, and

SR 3.3.1.2.3 b) 24 hours in CONDITION 3 or 4.

SR 3.3.1.2.7 2. CHANNEL CALIBRATION** at least once per 18 (24 — LE.1) months. ← Add proposed Note 2 to SR 3.3.1.2.7 — L.3

SR 3.3.2.1.6 b. Performance of a CHANNEL FUNCTIONAL TEST: ← Add proposed SNR determination — M.1

~~1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and~~ — L.4

SR 3.3.1.2.6 2. At least once per 31 days. — Add proposed Note to SR 3.3.1.2.6 — L3

SR3.3.1.2.4 c. Verifying, ~~prior to withdrawal of control rods~~, that the SRM count rate is at least 0.7 cps# (M.2) ~~with the detector fully inserted~~ — LA.1

Table 3.3.1.2-1 Note (a) { *With IRM's on range 2 or below.

Note 1 to SR 3.3.1.2.7 { **Neutron detectors may be excluded from CHANNEL CALIBRATION.

SR 3.3.1.2.4 { #Provided signals-to-noise ratio is ≥ 2 (20:1 — M.6). Otherwise, 3 cps.

R.1

CTS 3/4.3.7.11

~~EXPLOSIVE GAS MONITORING INSTRUMENTATION~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.3.7.11 The explosive gas monitoring instrumentation channels shown in Table 3.3.7.11-1 shall be OPERABLE with their Alarm/Trip setpoints set to ensure that the limits of specification 3.11.2.1 are not exceeded.~~

~~APPLICABILITY: During operation of the main condenser air ejector.~~

~~ACTION:~~

a. ~~With an explosive gas monitoring instrumentation channel Alarm/Trip setpoint less conservative than required by the above specification, declare the channel inoperable, and take the ACTION shown in Table 3.3.7.11-1.~~

b. ~~With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.11-1. Restore the inoperable instrumentation channels to an OPERABLE status within 30 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C. within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.~~

c. ~~The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.7.11 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies shown in Table 4.3.7.11-1.~~

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CTS 3/4.3.7.11

R.1

TABLE 3.3.7.11-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)		
a. Hydrogen Monitor	1/train	110

TABLE NOTATION

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours. If the recombiner(s) temperature remains constant and THERMAL POWER has not changed, the grab sample collection frequency may be changed to 8 hours.

R.1

CTS 3/4.3.7.11

TABLE 4.3.7.11-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION*	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	D	M	Q	**

TABLE NOTATION

* The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

** During operation of the main condenser air ejector.

page 6 of 6

R.1

CTS 3/4.3.7.12

~~LOOSE-PART DETECTION SYSTEM~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.3.7.12 The loose-part detection system shall be OPERABLE.~~

~~APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.~~

~~ACTION:~~

~~a. With one or more loose-part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.6.c within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.7.12 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:~~

~~a. CHANNEL CHECK at least once per 24 hours,~~

~~b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~c. CHANNEL CALIBRATION at least once per 18 months.~~

Page 2 of 2

A.1

ITS 3.3.2.2

INSTRUMENTATION

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.2.2 3.3.8 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.8-1 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.8-2.~~ LA.1

APPLICABILITY: ~~OPERATIONAL CONDITION 1.~~ THERMAL POWER ≥ 25% RTP L.1

ACTION: add proposed ACTIONS Note A.2

ACTIONS A and B a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable until the channel is restored to OPERABLE status ~~with its trip setpoint adjusted consistent with the Trip Setpoint value.~~ LA.1

b. With one or more channels required by Table 3.3.8-1 inoperable: L.2

ACTION B 1. Within 2 hours, verify sufficient channels remain OPERABLE or tripped* to maintain trip capability, and add proposed Required Action C.1

ACTION A 2. Within 7 days, either place the inoperable channel(s) in the trip system in the tripped* condition ~~or restore the inoperable channel(s) to OPERABLE status.~~ A.3

ACTION C c. Otherwise, be in at least ~~STARTUP~~ < 25% RTP within ~~6~~ 4 hours. L.1

SURVEILLANCE REQUIREMENTS

SR 3.3.2.2.1 through SR 3.3.2.2.3 4.3.8.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.8.1-1.

SR 3.3.2.2.4 4.3.8.2 LOGIC SYSTEM FUNCTIONAL TESTS and ~~simulated automatic operation~~ breaker and valve actuation of all channels shall be performed at least once per ~~18~~ 24 months. A.4 LD.1

LA.2

~~*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur.~~

TABLE 3.3.8-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LCO 3.3.2.2

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level-High, Level 8	4*

A.1

ITS 3.3.2.2

Page 6 of 8

Note to Surveillance Requirements

---

*A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition.

TABLE 3.3.8-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

SR 3.3.2.2.3

TRIP FUNCTION	~~TRIP SETPOINT~~	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High, Level 8	~~< 55.5 inches*~~	< 56.0 inches*

LCO 3.3.2.2

LA.1

A.5

LF.1

A.1

~~*See Bases Figure B 3/4 3-1.~~

A.5

ITS 3.3.2.2

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TABL. 3.8.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	SR 3.3.2.2.1 CHANNEL CHECK	SR 3.3.2.2.2 CHANNEL FUNCTIONAL TEST	SR 3.3.2.2.3 CHANNEL CALIBRATION
LCO 3.3.2.2 a. Reactor Vessel WAter Level-High, Level 8	S	Q	~~R~~ 24 months LE.1

A.1

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ITS 3.3.2.2

ITS 3.4.1

3/4.4 REACTOR COOLANT SYSTEM

A.1

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

A.2

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 **3.4.1.1** Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

within Region III of Figure 3.4.1-1 A.3

ACTION

LCO 3.4.1 a. With only one (1) reactor coolant system recirculation loop in operation, ~~comply with Specification 3.4.1.5~~ and: L.1

ACTION G 1. Within ~~four (4)~~ 12 hours: satisfy the requirements of the LCO A.2

~~a) Place the recirculation flow control system in the Master Manual mode or lower, and~~ LA.1

~~b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.2, and~~ A.4

LCO 3.4.1

as specified in the COLR for Single Loop Operation LA.6

c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation ~~by 0.01~~ per Specification 3.2.3, and,

d) Reduce the Average Power Range Monitor (APRM) Scram ~~and Rod Block~~ and Rod Block Monitor ~~Trip Setpoints~~ and Allowable Values to those applicable to single recirculation loop operation per Specifications 2.2.1 and 3.3.6. A.5

e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.

ACTION H 2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.

ACTION D

b. With no reactor coolant recirculation loops in operation:

~~1. Take the ACTION required by Specification 3.4.1.5, and~~ A.3

2. Be in at least HOT SHUTDOWN within ~~the next six (6)~~ 12 hours. L.2

Page 6 of 10

A.1 ITS 3.4.2

REACTOR COOLANT SYSTEM

Add proposed LCO 3.4.2, Applicability, and ACTIONS A.2

SURVEILLANCE REQUIREMENTS

4.4.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by: 24 LD.1

SR 3.4.2.1 a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic power units, and

SR 3.4.2.2 b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and
2. Less than or equal to 11% of stroke per second closing.

A.1 ITS 3.4.3

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

LCO 3.4.3 3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION A With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

L.1 add proposed NOTE 1 to SR 3.4.3.1

add proposed NOTE 2 to SR 3.4.3.1

SR 3.4.3.1 4.4.1.2.1 Each of the above required jet pumps shall be demonstrated OPERABLE ~~prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and~~ at least once per 24 hours by measuring and recording each of the below specified parameters and verifying that no two of the following conditions occur ~~when both recirculation loops are operating with balanced flow.~~ LA.1

a. The indicated recirculation loop drive A.2 flow differs by more than 10% from the established flow control valve position ~~loop flow characteristics for two recirculation loop operation.~~ LA.1 L.2

b. The indicated total core flow differs by more than 10% from the established calculated A.3 total core flow value ~~derived from either the:~~

~~1. Established THERMAL POWER-core flow relationship, or~~

~~2. Established core plate differential pressure-core flow relationship for two recirculation loop operation.~~ LA.2

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established two recirculation loop operation patterns by more than ~~10%~~ 20% L.4

L.1 add proposed NOTE 1 to SR 3.4.3.1

add proposed NOTE 2 to SR 3.4.3.1

SR 3.4.3.1 4.4.1.2.2 During single recirculation loop operation, each of the above ~~required~~ in the operating loop jet pumps shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur: L.3

a. The indicated recirculation loop drive A.2 flow in the operating loop differs by more than 10% from the established ~~single~~ recirculation flow control valve position ~~loop flow characteristics.~~ LA.1 L.2

b. The indicated calculated A.3 total core flow differs by more than 10% from the established total core flow value ~~from single recirculation loop flow measurements derived from either the:~~

~~1. Established THERMAL POWER-core flow relationship, or~~

~~2. Established core plate differential pressure-core flow relationship for two recirculation loop operation.~~ LA.2

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop by more than ~~10%~~ 20% L.4

A.1

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

jet pump A.6

LCO 3.4.1 **3.4.1.3** Recirculation loop flow mismatch shall be maintained within:

SR 3.4.1.1

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 during two recirculation loop operation.

ACTION:

Action F

With recirculation loop flows different by more than the specified limits, either:

a. ~~Restore the recirculation loop flows to within the specified limit within 2 hours, or~~ A.7

b. Declare the recirculation loop with the lower flow not in operation. ~~and take the ACTION required by Specification 3.4.1.1.~~ A.14

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.1.1 Note L.3

SR 3.4.1.1 **4.4.1.3** Recirculation loop flows shall be verified to be within the limits at least once per 24 hours.

jet pump A.6

A.1 ITS 3.4.11

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

LA.3

LCO 3.4.11
SR 3.4.11.3

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel ~~steam space~~ coolant and the bottom ~~head drain~~ line coolant is less than or equal to 145°F, and:

A.7

SR 3.4.11.4

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle ~~and operating~~ recirculation loops is less than or equal to 50°F ~~and the operating loop flow rate is less than or equal to 50% of rated loop flow~~.

A.8

LA.4

Note to SR 3.4.11.3 and SR 3.4.11.4

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With temperature differences ~~and/or flow rates~~ exceeding the above limits, suspend startup of any idle recirculation loop.

add proposed ACTIONS A, B, and C

M.2

SURVEILLANCE REQUIREMENTS

SR 3.4.11.3
SR 3.4.11.4

4.4.1.4 The temperature differentials ~~and flow rate~~ shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

LA.4

REACTOR COOLANT SYSTEM

A.1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

THERMAL HYDRAULIC STABILITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.1

3.4.1.5 Forced core circulation shall be maintained with:

~~a. Total core flow greater than or equal to 45% of rated core flow, or~~ A.8

b. THERMAL POWER within Region III of Figure 3.4.1.5-1, or

ACTION A

c. THERMAL POWER within Region II of Figure 3.4.1.5-1 AND APRM and LPRM noise levels not exceeding the larger of: i) Three (3) times the established baseline noise levels or, ii) 10% peak-to-peak indicated noise level. A.9

APPLICABILITY: OPERATIONAL CONDITION 1 ← and 2 A.13

ACTION

ACTION C

a. In Region I of Figure 3.4.1.5-1:

1. With at least 1 reactor coolant recirculation loop in operation ~~immediately initiate action to:~~

   ~~a) Decrease THERMAL POWER by control rod insertion, completing the power decrease~~ within two (2) hours ~~to~~ exit Region I or,

   ~~b) Increase core flow with the operating Recirculation Loop(s), to~~ exit Region I within two (2) hours.

   LA.2

2. With no reactor coolant recirculation loops in operation: LA.2

   ACTION D

   a) ~~Immediately~~ reduce CORE THERMAL POWER ~~by inserting control rods, observing the~~ indicated APRM and LPRM noise levels, and complete power reduction to below 36% of RATED CORE THERMAL POWER within two (2) hours, and A.10

   ACTION E

   b) If indicated LPRM or APRM noise levels exceed 10% peak-to-peak, immediately place the reactor mode switch in the SHUTDOWN position.

   Required Action D.3

   c) ~~Comply with Specification 3.4.1.1 ACTION b.2~~ A.3

   Be in MODE 3 in 12 hours

   L.2

ITS 3.4.1

REACTOR COOLANT SYSTEM A.1

ACTION (Continued)

ACTION B

b. In Region II of Figure 3.4.1~~.5~~-1, with APRM or LPRM neutron flux noise levels exceeding the larger of: i) Three (3) times the established baseline noise levels, or ii) 10% peak-to-peak noise indication. LA.3

1. ~~Immediately initiate corrective action by inserting control rods or increasing core flow to restore the noise levels to within the required limit within 2 hours, otherwise.~~ LA.3

2. ~~Insert control rods to reduce THERMAL POWER and/or increase core flow to~~ enter Region III of Figure 3.4.1~~.5~~-1 within the ~~next~~ 2 hours. M.2

Add ACTION E M.3

SURVEILLANCE REQUIREMENTS

ACTION A

4.4.1.5 When operating within Region II of Figure 3.4.1.5-1, verify:

1. That the APRM and LPRM neutron flux noise levels do not exceed the larger of: i) Three (3) times the established baseline levels or, ii) 10% peak-to-peak indicated noise level:

   a. At least once per 12 hours, and

   b. ~~Initiate the surveillance within 15 minutes~~ after entering the region or completing an increase of at least 5% of RATED THERMAL POWER, ~~completing the surveillance~~ within ~~the next 30~~ 45 minutes. LA.4 A.12

2. That ~~core flow is greater than or equal to 39% of rated core flow~~ operation is not within Region I of Figure 3.4.1-1 at least once per 12 hours. A.9

Add proposed SR 3.4.1.2 M.4

~~#Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center region of the core should be monitored.~~ LA.5

A.1

Figure 3.4.1-1 (Page 1 of 1)
Power Versus Flow

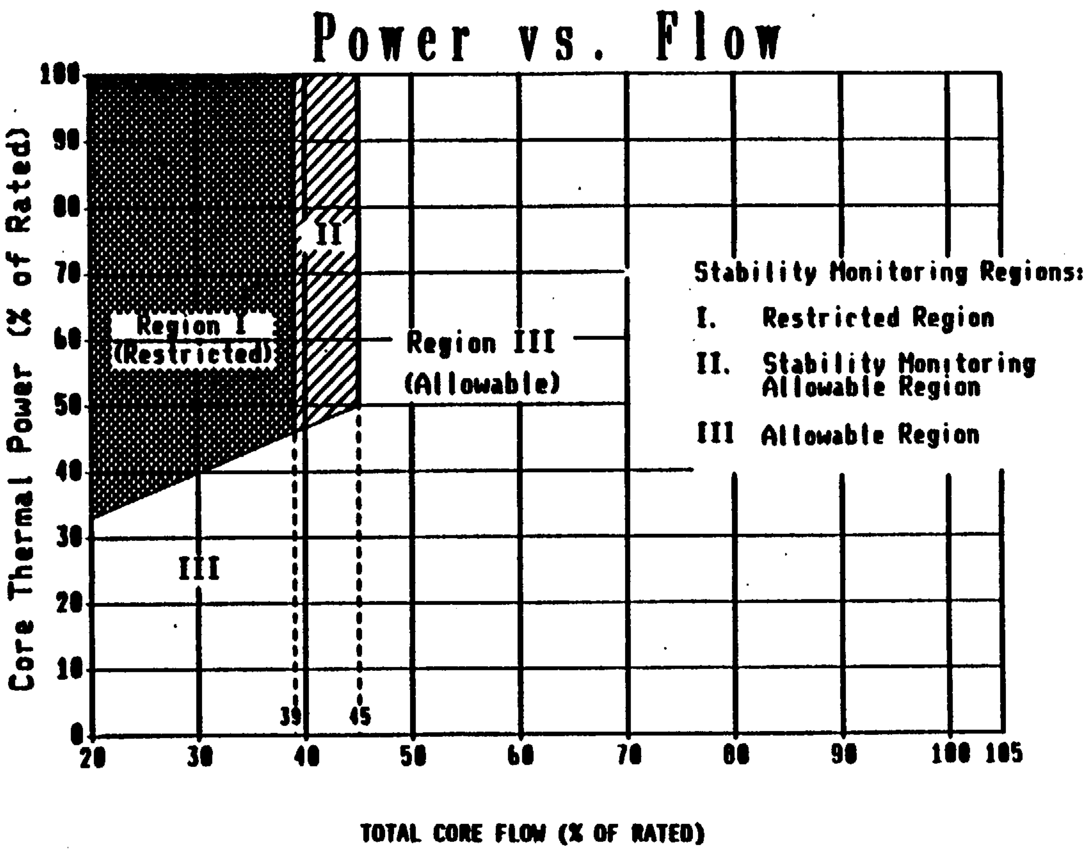

Figure 3.4.1.5-1

ITS 3.4.1

ITS 3.3.3.1

REACTOR COOLANT SYSTEM

A.1

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 12 of the below listed 13 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed ~~with OPERABLE position indication~~.

R.1

- a. 2 safety/relief valves @1205 psig ±3%
- b. 3 safety/relief valves @1195 psig ±3%
- c. 2 safety/relief valves @1185 psig ±3%
- d. 4 safety/relief valves @1175 psig ±3%
- e. 2 safety/relief valves @1150 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

~~b. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

SURVEILLANCE REQUIREMENTS

R.1

~~4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:~~

~~a. CHANNEL CHECK at least once per 31 days, and a~~
~~b. CHANNEL CALIBRATION at least once per 18 months.**~~

4.4.2.2 The low low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

---

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Following testing, lift settings shall be within ±1%.

#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.

~~**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.~~

R.1

<See ITS 3.4.4>

page 10 of 10

A.1

ITS 3.3.5.1

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 12 of the below listed 13 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

a.	2	safety/relief valves @1205 psig ±3%
b.	3	safety/relief valves @1195 psig ±3%
c.	2	safety/relief valves @1185 psig ±3%
d.	4	safety/relief valves @1175 psig ±3%
e.	2	safety/relief valves @1150 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The safety/relief valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months.**

L.5

4.4.2.2 The low low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months.

---

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Following testing, lift settings shall be within ±1%.

#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

(See ITS 3.4.4)

ITS 3.4.4

A.1

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.4.4 3.4.2 The safety valve function of 12 of the below listed 13 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed ~~with OPERABLE position indication~~. L.1 ⟨Moved to ITS 3.3.3.1⟩ A.2 L.1

- a. 2 safety/relief valves @1205 psig ±3%
- b. 3 safety/relief valves @1195 psig ±3%
- c. 2 safety/relief valves @1185 psig ±3%
- d. 4 safety/relief valves @1175 psig ±3%
- e. 2 safety/relief valves @1150 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Action A a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A.2

SURVEILLANCE REQUIREMENTS

~~4.4.2.1 The safety/relief~~ valve stem position indicators of each safety/relief valve shall be demonstrated OPERABLE by performance of a:

- a. CHANNEL CHECK at least once per 31 days, and a
- b. CHANNEL CALIBRATION at least once per 18 months.**

⟨Moved to ITS 3.3.3.1⟩ A.2

4.4.2.2 The low low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months. L.2 A.3 ⟨Moved to ITS 3.3.5.1⟩

SR 3.4.4.1 *The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Following testing, lift settings shall be within ±1%. LA.1 A.4

Note to SR3.4.4.1 #~~Up to two inoperable valves may be~~ replaced with spare OPERABLE valves with lower setpoints ~~until the next refueling outage~~. L.3

**The ~~provisions of Specification 4.0.4 are not~~ applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. A.2

⟨Moved to ITS 3.3.3.1⟩

Page 2 of 2

ITS 3.4.7

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE
LEAKAGE DETECTION SYSTEMS

A.1

LIMITING CONDITION FOR OPERATION

LCO 3.4.7

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere particulate radioactivity monitoring system, L.1

b. The primary containment sump flow monitoring system, and

c. Either the primary containment air coolers condensate flow rate monitoring system or the primary containment atmosphere gaseous radioactivity monitoring system. L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION: add proposed Note to Actions A and D L.2

Actions A, B, C, and D

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; L.1

Action E

otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

add proposed ACTION F A.2

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system detection systems shall be demonstrated OPERABLE by: add proposed Note to SR Table L.3

SR 3.4.7.1
SR 3.4.7.2
SR 3.4.7.3

a. Primary containment atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per ~~18~~ 24 months. L.1 LE.1

SR 3.4.7.2
SR 3.4.7.3

b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per ~~18~~ 24 months. LE.1

SR 3.4.7.2
SR 3.4.7.3

c. Primary containment air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per ~~18~~ 24 months. LE.1

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

A.1

ITS 3.4.5

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

LCO 3.4.5

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24 hour period. the previous A.2

d. 1 gpm leakage at a reactor coolant system pressure at 1000 ± 50 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1. A.3 <Moved to LCO 3.4.6>

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24 hour period. the previous A.2 in Mode 1. L.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

ACTION C a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION A b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and
ACTION C in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limits, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A.3 <Moved to LCO 3.4.6>

d. With one or more high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours by local indication; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 12 hours.

ACTION B e. With any reactor coolant system leakage greater than the limit in e, above, identify the source of leakage within 4 hours or be in at
ACTION C least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

is not IGSCC susceptible material M.1

or reduce leakage to within limit A.4

Page 3 of 4

A.1

ITS 3.4.6

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

L.4

0.5 gpm leakage per nominal inch of valve size up to a maximum leakage of 5 gpm for each PIV

LCO 3.4.6 3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24 hour period.

SR 3.4.6.1 LCO 3.4.6 d. 1 gpm leakage at a reactor coolant system pressure at 1000 ± 50 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

LA.1

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24 hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3. ← add proposed MODE 3 RHR Allowance L.1

ACTION: ← add proposed ACTIONS Notes 1 and 2 A.2

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limits, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION A

ACTION B

L.2

d. ~~With one or more high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours by local indication; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 12 hours.~~

LC.1

e. With any reactor coolant system leakage greater than the limit in e, above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

<See ITS 3.4.5>

M.1

add Required Actions A.1 and A.2 Note

A.1

ITS 3.4.5

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1 4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within* each of the above limits ~~on average~~ once per ~~8 hours not to exceed~~ 12 hours.

L.1

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE:

a. Pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

1. At least once per 18 months, and
2. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

b. By demonstrating OPERABILITY of the high/low pressure interface valve leakage pressure monitors by performance of a:

1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
2. CHANNEL CALIBRATION at least once per 18 months,

With the alarm setpoint for the:

1. HPCS system ≤ 100 psig.
2. LPCS system ≤ 500 psig.
3. LPCI/shutdown cooling system ≤ 400 psig.
4. RHR shutdown cooling ≤ 190 psig.
5. RCIC ≤ 90 psig.

A.3

⟨Moved to LCO 3.4.6⟩

~~*Technical Specification 4.0.2 does not apply.~~ L.1

Page 4 of 4

A.1

ITS 3.4.6

SURVEILLANCE REQUIREMENTS

<See ITS 3.4.5>

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits on average once per 8 hours not to exceed 12 hours.*

SR3.4.6.1 4.4.3.2.2 Each reactor coolant system pressure isolation valve ~~specified in Table 3.4.3.2-1~~ shall be demonstrated OPERABLE: LA.1

a. Pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

1. At least ~~once per 18 months, and~~ LA.2
2. ~~Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.~~ L.3

Note to SR 3.4.6.1 The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

~~b. By demonstrating OPERABILITY of the high/low pressure interface valve leakage pressure monitors by performance of a:~~

~~1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~2. CHANNEL CALIBRATION at least once per 18 months,~~

~~With the alarm setpoint for the:~~

~~1. HPCS system ≤ 100 psig.~~

~~2. LPCS system ≤ 500 psig.~~

~~3. LPCI/shutdown cooling system ≤ 400 psig.~~

~~4. RHR shutdown cooling ≤ 190 psig.~~

~~5. RCIC ≤ 90 psig.~~

LC.1

<See ITS 3.4.5>

---

*Technical Specification 4.0.2 does not apply.

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A.1 ITS 3.4.6

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SYSTEM	VALVE NUMBER	FUNCTION
a. LPCS	E21-F006	LPCS Injection
	E21-F005	LPCS Injection
b. HPCS	E22-F005	HPCS Injection
	E22-F004	HPCS Injection
c. RHR	E12-F041A	LPCI Injection
	E12-F041B	LPCI Injection
	E12-F041C	LPCI Injection
	E12-F042A	LPCI Injection
	E12-F042B	LPCI Injection
	E12-F042C	LPCI Injection
	E12-F050A	Shutdown Cooling Return
	E12-F050B	Shutdown Cooling Return
	E12-F053A	Shutdown Cooling Return
	E12-F053B	Shutdown Cooling Return
	E12-F009	Shutdown Cooling Suction
	E12-F008	Shutdown Cooling Suction
d. RCIC	E51-F066	RCIC Head Spray
	E5I-F065	RCIC Head Spray

LA.1

Page 6 of 6

ITS 3.4.8

A.1

REACTOR COOLANT SYSTEM

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.8 3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcurie per gram DOSE EQUIVALENT I-131, ~~and~~

~~b. Less than or equal to 100/E microcuries per gram.~~ L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2, 3, ~~and 4.~~ with any main steam line not isolated

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the specific activity of the primary coolant; L.2

add proposed Note A to Action A L3

ACTION A

ACTION B

1. Greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 but less than or equal to 4 microcurie per gram DOSE EQUIVALENT I-131 ~~for more than 48 hours during one continuous time interval~~ or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, ~~be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.~~ L.2

~~2. Greater than 100/E microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.~~ L.1

Required Actions A.1 and B.1

b. In OPERATIONAL CONDITION 1, 2, 3 ~~or 4~~ with the specific activity of the primary coolant greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 ~~or greater than 100/E microcuries per gram~~, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within the limit. L.2 L.1

~~c. In OPERATIONAL CONDITION 1 or 2, with:~~

~~1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in 1 hour*, or~~

~~2. The off-gas level, prior to the holdup line, increased by more than 25,000 microcuries per second in one hour during steady state operation at release rates less than 100,000 microcuries per second, or~~ A.2

~~*Not applicable during the Startup Test Program.~~

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ITS 3.4.8

A.1

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

A.2

~~3. The off-gas level, prior to the holdup line, increased by more than 15% in 1 hour during steady state operation at release rates greater than 100,000 microcuries per second,~~

~~perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

Page 5 of 6

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED	
~~1. Gross Beta and Gamma Activity Determination~~	~~At least once per 72 hours~~	~~1, 2, 3~~	L.1
SR 3.4.8.1 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days 7	1	M.1
~~3. Radiochemical for Ē Determination~~	~~At least once per 6 months*~~	~~1~~	L.1
4. Isotopic Analysis for Iodine Including I-131, I-133 and I-135	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ~~ACTION b.~~ Required Actions A.1 and B.1	1#, 2#, 3#, 4#	L.2
	~~b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.~~	~~1, 2~~	A.2
~~5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88~~	~~At least once per 31 days~~	~~1~~	LA.1

~~*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.~~ L.1

#Until ~~the specific activity of the primary coolant system is restored to within its limits.~~ Required Actions A.1 and B.1

A.1

ITS 3.4.8

Page 6 of 6

A.1

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

A.1 within limits

LCO 3.4.11 3.4.6.1 The reactor coolant system temperature and pressure shall be ~~limited in accordance with the limit lines shown on Figure~~~~s~~ 3.4.6.1-1 ~~and 3.4.6.1-1a:~~ ~~(1) curves A~~ for hydrostatic or leak testing; ~~(2) curves B~~ for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and ~~(3) curves C~~ for operations with a critical core other than low power PHYSICS TESTS, with: Figure 3.4.6.1-1a Figure 3.4.6.1-1b

SR 3.4.11.1

SR 3.4.11.2

SR 3.4.11.1
a. A maximum heatup of 100°F in any one hour period,
b. A maximum cooldown of 100°F in any one hour period,
c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

SR 3.4.11.5 SR 3.4.11.6 SR 3.4.11.7
d. The reactor vessel flange and head flange temperature greater than or equal to 86°F when reactor vessel head bolting studs are under tension.

<LAR297> A.9

APPLICABILITY: At all times.~~*~~

ACTION: add proposed Conditions A and C Notes A.2

A.3 M.1 add Required Actions A.2 and C.2 Completion times

ACTIONS A and C With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; ~~perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system;~~ ~~determine that the reactor coolant system remains acceptable for continued operations~~ or be in at least HOT SHUTDOWN

ACTION B within 12 hours and in COLD SHUTDOWN within the following 24 hours.

LA.2

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1 4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits ~~and to the right of the limit lines~~ of Figures 3.4.6.1-1 ~~and 3.4.6.1-1a curves A or B~~, as applicable, at least once per 30 minutes. 3.4.6.1-1a, and 3.4.6.1-1b

LA.1

~~*During shutdown conditions for hydrostatic or leak testing or heatup by nonnuclear means, the average coolant temperature limit of Table 1.2 for Cold Shutdown and Hot Shutdown may be increased to 212°F.~~

<LAR297> A.9

A.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

within LA.1

SR 3.4.11.2 4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be ~~to the right of~~ the criticality limit line of Figures ~~3.4.6.1-1 and 3.4.6.1-1a curves C~~ 3.4.6.1-1b within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

<LAR297> A.9

~~4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1 and 3.4.6.1-1a.~~ ② and 3.4.6.1-1b A.4

SR 3.4.11.5
SR 3.4.11.6
SR 3.4.11.7 4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 86°F:

a. In OPERATIONAL CONDITION 4 when the reactor coolant temperature is:

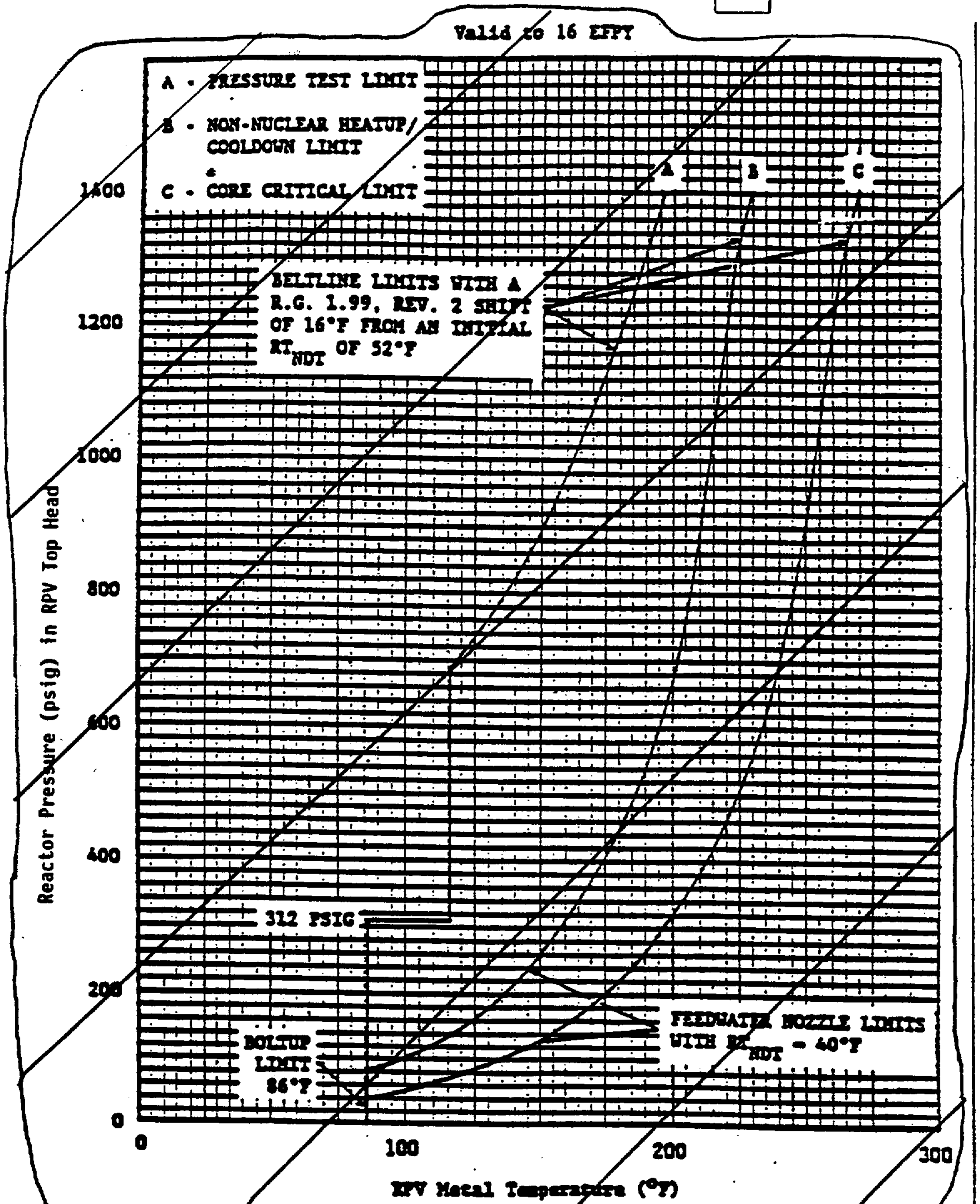
SR 3.4.11.7 1. ≤ 106°F, at least once per 12 hours. ← add proposed SR 3.4.11.7 Note

SR 3.4.11.6 2. ≤ 91°F, at least once per 30 minutes. add proposed SR 3.4.11.6 Note A.5

SR 3.4.11.5 b. ~~Within 30 minutes prior to and~~ at least once per 30 minutes during tensioning of the reactor vessel head bolting studs. A.6

A.1

ITS 3.4.11



Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure

Figure 3.4.6.1-1

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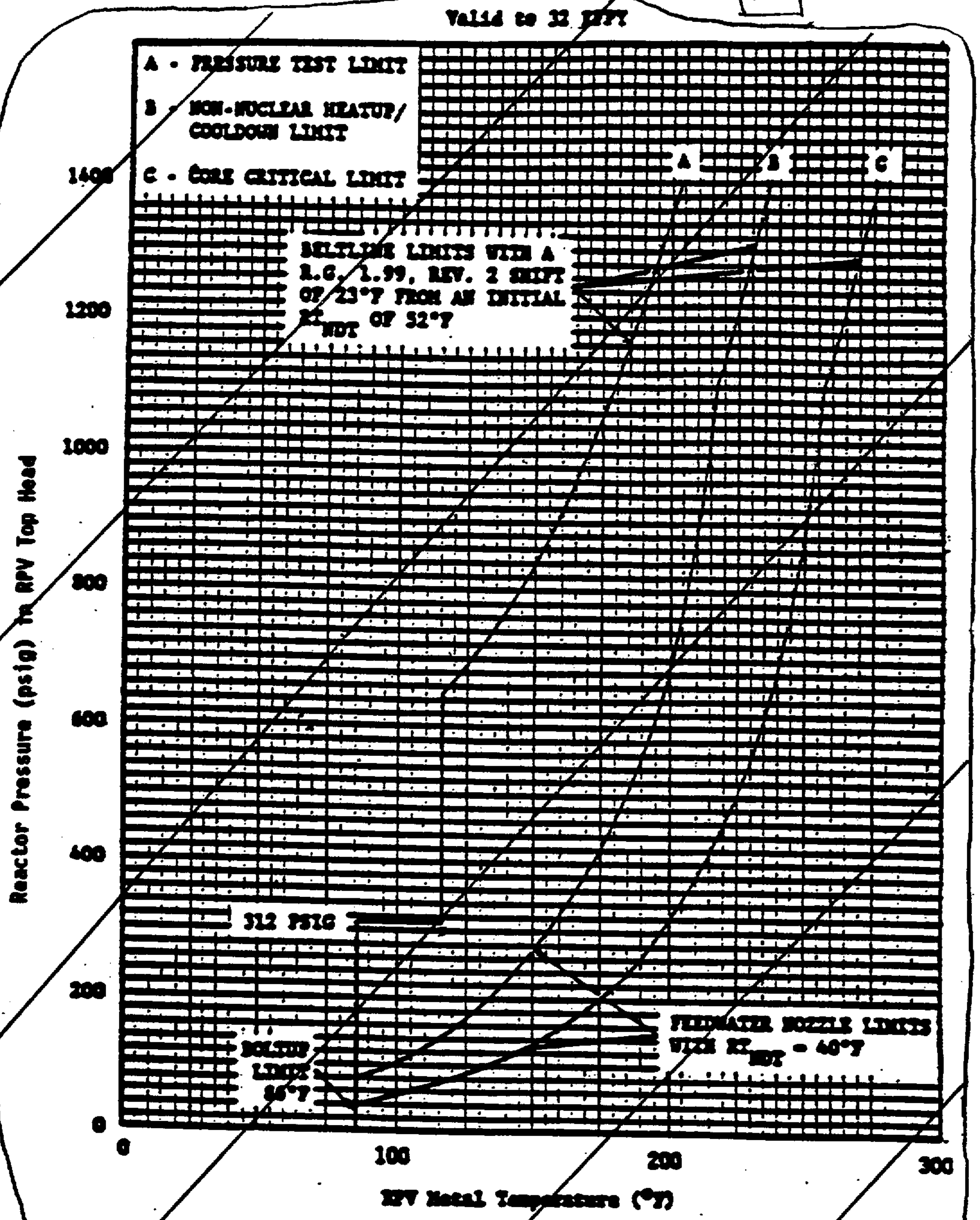
A.1



Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure

Figure 3.4.6.1-1a

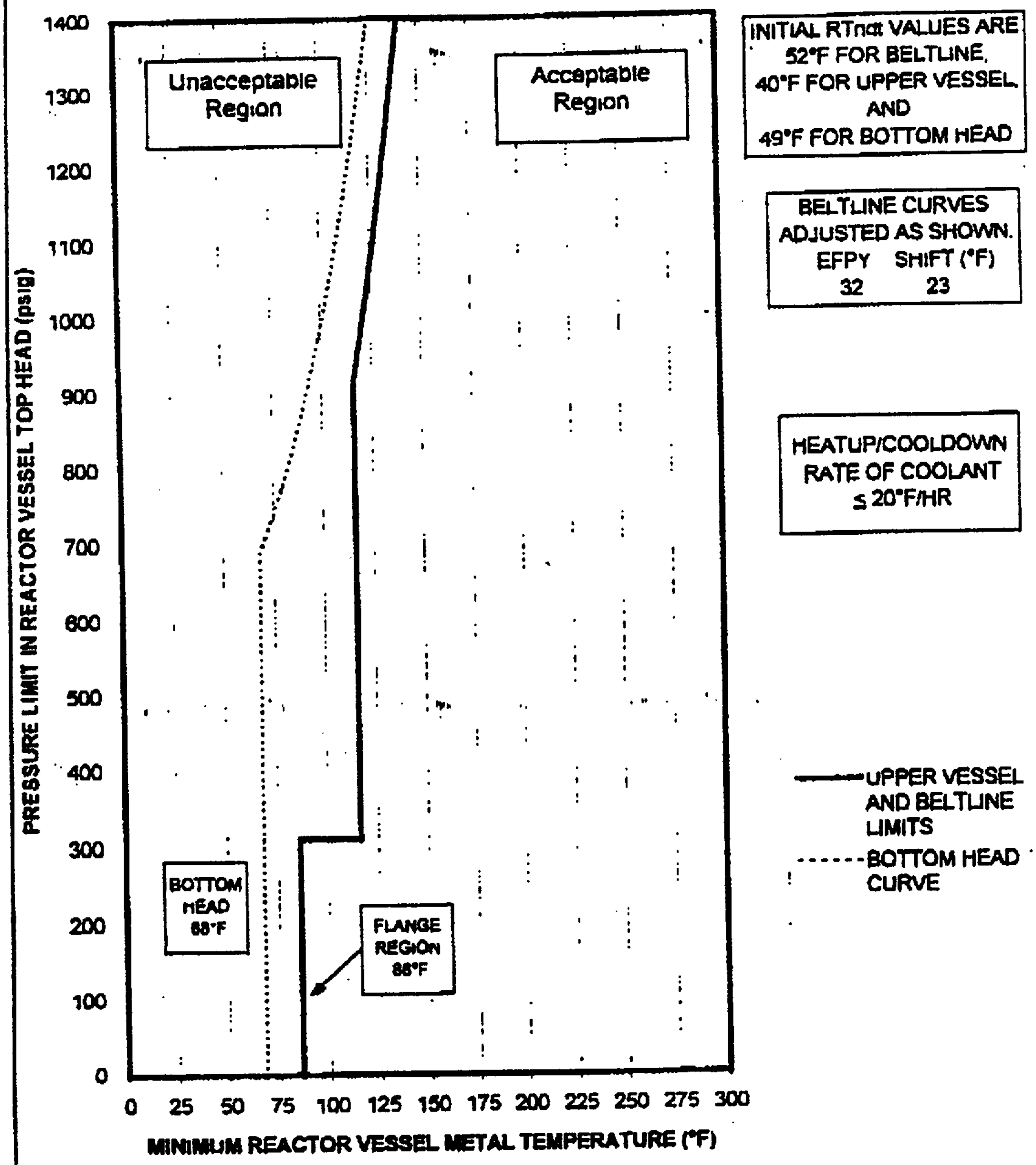
Replace with Figures 3.4.6.1-4, 3.4.6.1-5, and 3.4.6.1-6

<LAR297>

A.9

A.1

ITS 3.4.11



P-T Curve for Hydrostatic or Leak Testing

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1

<LAR297> A.9

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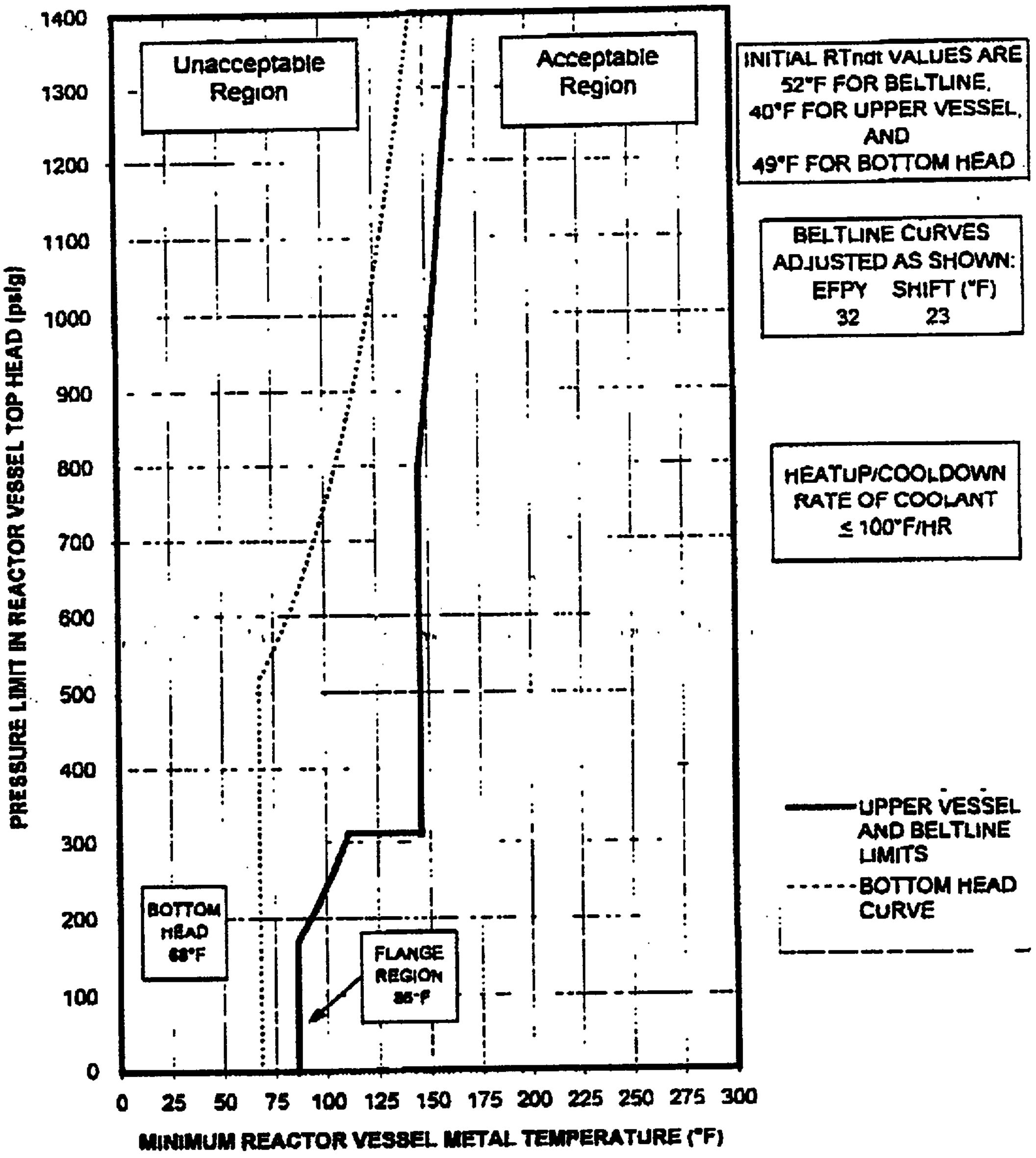


P-T Curve for Heatup by Non-Nuclear Means,
Cooldown Following a Nuclear Shutdown and Low Power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

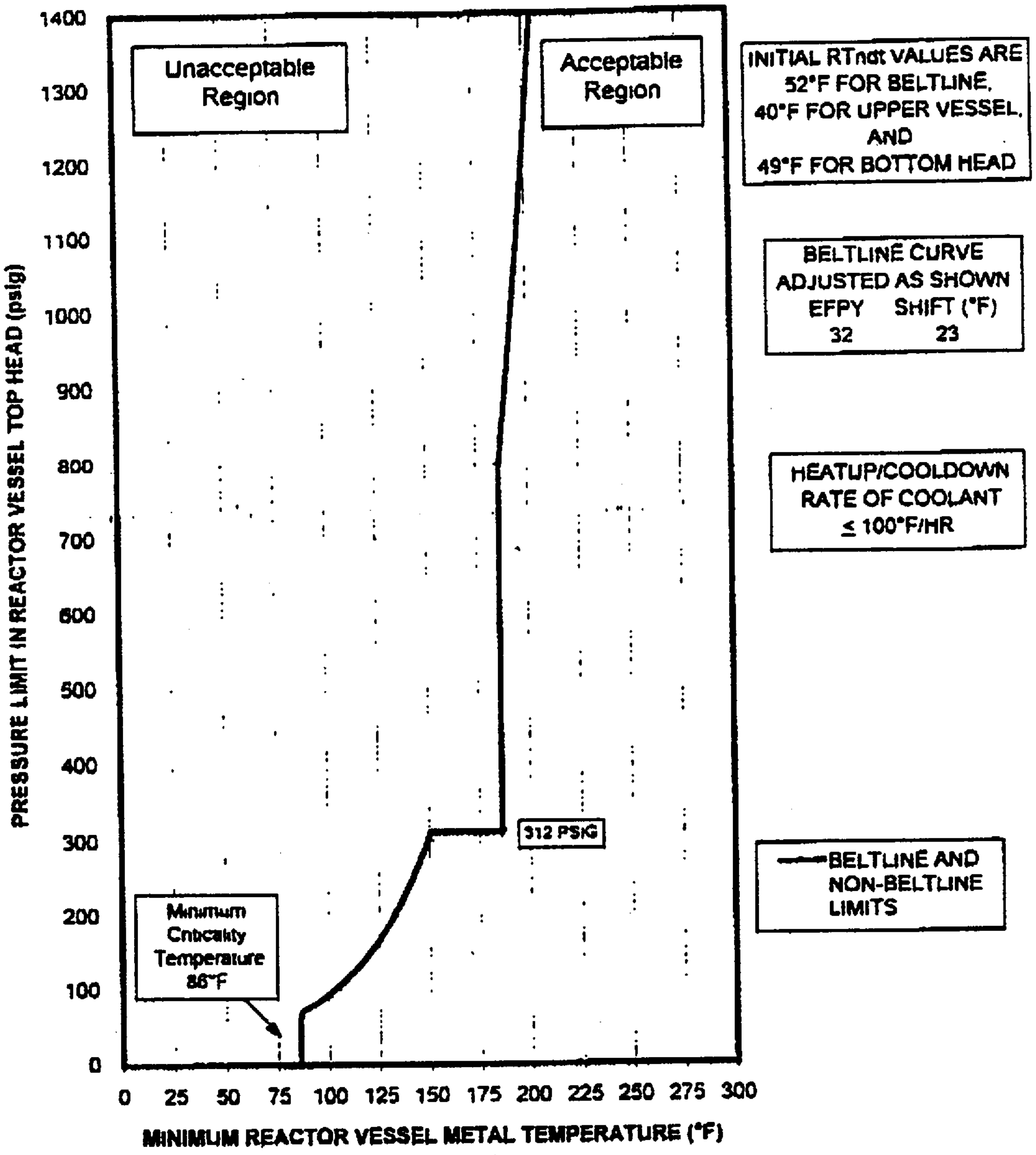
Valid to 32 EFPY

Figure 3 4.6 1-1a

<LAR297> A.9

A.1

ITS 3.4.11



P-T Curve for Operation with a Critical Core other than low power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1b

<LAR297> A9

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

SPECIMEN HOLDER*	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFFECTIVE FULL POWER YEARS)
Capsule 1	300°	0.6	6
Capsule 2	120°	0.6	15
Capsule 3	30°	0.6	Spare
Neutron Dosimeter	30°	-	1st Refueling Outage

---

*Each capsule includes an Fe, Ni, and Cu flux wire. The neutron dosimeter contains three Cu and three Fe flux wires.

A.1

ITS 3.4.11

<LAR297>

A.9

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I.TS 3.4.12

A.1

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

LCO 3.4.12 **3.4.6.2** The pressure in the reactor steam dome shall be less than or equal to 1020 psig. L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1~~*~~ and 2~~*~~. M.1

ACTION:

ACTION A — With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes

ACTION B — or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 **4.4.6.2** The reactor steam dome pressure shall be verified to be less than or equal to 1020 psig at least once per 12 hours. L.1

M.1

~~*Not applicable during anticipated transients.~~

A.1

ITS 3.6.1.3

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

A.7

LCO 3.6.1.3 3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or

SR 3.6.1.3.6 equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

add proposed Note 1 to ACTIONS — L.5

ACTION:

add proposed Note 2 to ACTIONS — A.2

With one or more MSIVs inoperable:

ACTION A 1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours either: — A.4

a) Restore the inoperable valve(s) to OPERABLE status, or — A.5

b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position. — L.2

ACTION E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

add proposed ACTION B — L.3

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

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Current Specification 3/4.4.8

R.1

REACTOR COOLANT SYSTEM

~~3/4.4.8 STRUCTURAL INTEGRITY~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.~~

~~APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.~~

~~ACTION:~~

a. ~~With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.~~

b. ~~With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.~~

c. ~~With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.8 No additional Surveillance Requirements other than those required by Specification 4.0.5.~~

Page 2 of 2

A.1

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

, unless at least one recirculation pump is in operation, L.1

LCO 3.4.9 3.4.9.1 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one shutdown cooling mode loop shall be in operation*~~##~~ ~~with each loop consisting of at least:~~ A.2

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

add proposed Actions Note 1. L.2

add proposed Actions Note 2 A.3

ACTION A

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within 1 hour ~~and at least once per 24 hours thereafter~~, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.~~**~~ A.4 A.5

ACTION B

b. With no RHR shutdown cooling mode loop or recirculation loop in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour. L.1

SURVEILLANCE REQUIREMENTS

add proposed SR 3.4.9.1 Note L.2

SR 3.4.9.1

Required Action B.2

4.4.9.1 At least one shutdown cooling mode loop of the residual heat removal system ~~or alternate method~~ shall be determined to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA.2

or recirculation loop L.1

LCO Note 2

#One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing ~~provided the other loop is OPERABLE and in operation.~~

LCO Note 1

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period ~~provided the other loop is OPERABLE.~~ L.3

~~##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.~~ A.2

~~**Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~ A.5

ITS 3.4.10

A.1

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

unless at least one recirculation pump is in operation L.1

LCO 3.4.10 3.4.9.2 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE* and at least one shutdown cooling mode loop shall be in operation** ## ~~with each loop consisting of at least:~~

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~

LA.1

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION: add proposed ACTIONS NOTE A.3

Action A a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

or recirculation loop L.1

Action B b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

or recirculation loop L.1

SR 3.4.10.1 4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system (or alternate method) shall be determined to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA.2

Required Action B.1

LCO Note 3 #One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing ~~provided the other loop is OPERABLE and in operation~~. L.2

~~*The normal or emergency power source may be inoperable.~~ A.4

LCO Note 2 **The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period ~~provided the other loop is OPERABLE~~. L.2

LCO Note 1 ##The shutdown cooling mode loop may be removed from operation during hydrostatic testing. L.3

ITS 3.5.1

A.1

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.5.1 3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~ LA.1

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.~~ LA.1

3. At least 6 OPERABLE~~**~~ ADS valves. A.6

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, ~~each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.~~ LA.1

2. At least 6 OPERABLE~~**~~ ADS valves. A.6

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 1, 2*~~#~~ and 3*. A.2

Add LCO Note L.3

APPL *The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~222~~ 150 psig. L.1

~~**See Specification 3.3.3 for trip system operability.~~ A.6

~~#See Special Test Exception 3.10.6.~~ A.2

ITS 3.5.1

A.1

EMERGENCY CORE COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

~~a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:~~ A.3

Action A
1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.
2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

Action C
3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

Action E
4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~b. For ECCS division 2, provided that ECCS divisions 1 and 3 are OPERABLE:~~ A.3

Action A
1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

Action C
2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

Action E
3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours~~*~~. A.4

Action B
c. For ECCS division 3, provided ~~that ECCS divisions 1 and 2 and~~ the RCIC system are OPERABLE: A.3 L.6
1. With ECCS division 3 inoperable, restore the inoperable division to OPERABLE status within 14 days.

Action E
2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~d. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE:~~ A.3

Action C
1. With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

~~*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~ A.4

ITS 3.5.1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

Action C — 2. With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

Action E — 3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours~~*~~. — A.4

~~e. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE and divisions 1 and 2 are otherwise OPERABLE:~~ — A.3

Action F — 1. With one of the above required ADS valves inoperable, restore the inoperable ADS valve to OPERABLE status within 14 days or be

Action G — in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ ~~122~~ 150 psig within the next 24 hours. — L.1

Action G — 2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ ~~122~~ 150 psig within the next 24 hours. — L.1

~~f. With an ECCS discharge line "keep filled" pressure alarm instrumentation channel inoperable, perform Surveillance Requirement 4.5.1.a.1 at least once per 24 hours.~~ — L.5

~~g. With an ECCS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or determine ECCS header delta P locally at least once per 12 hours; otherwise, declare the associated ECCS inoperable.~~ — L.5

~~h. With Surveillance Requirement 4.5.1.d.2 not performed at the required interval due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.~~ — M.1

Add proposed Action H — A.3

~~*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~ — A.4

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A.1 ITS 3.5.1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

~~i. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6.C within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~ L.2

~~j. With one or more ECCS corner room watertight doors inoperable, restore all the inoperable ECCS corner room watertight doors to OPERABLE status within 14 days, otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~ LA.3

k. With ADS accumulator backup compressed gas system bottle pressure
less than 500 psig, restore ADS accumulator backup compressed gas
ACTION D system bottle pressure to greater than 500 psig within 72 hours or
ACTION E declare the associated ADS valves inoperable, and follow Action e of
this specification.

A.1

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1 ECCS divisions 1, 2, and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI, and HPCS systems: LA.2

SR 3.5.1.1 — 1. Verifying ~~by venting at the high point vents~~ that the system piping from the pump discharge valve to the system isolation valve is filled with water.

~~2. Performance of a CHANNEL FUNCTIONAL TEST of the:~~

~~a) Discharge line "keep filled" pressure alarm instrumentation, and~~ L.5

~~b) Header delta P instrumentation.~~

SR 3.5.1.2 — 3. Verifying that each valve (manual, power-operated, or automatic,) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

~~4. Verifying that each ECCS corner room watertight door is closed, except during entry to and exit from the room.~~ LA.3

b. Verifying that, when tested pursuant to Specification 4.0.5, each:

SR 3.5.1.5 —

1. LPCS pump develops a flow of at least 6350 gpm against a test line pressure greater than or equal to 290 psig.
2. LPCI pump develops a flow of at least 7200 gpm against a test line pressure greater than or equal to 130 psig.
3. HPCS pump develops a flow of at least 6200 gpm against a test line pressure greater than or equal to 330 psig.

24 LD.1

c. For the LPCS, LPCI and HPCS systems, at least once per 18 months:

an actual or L.4

SR 3.5.1.6 — 1. Performing a system functional test which includes simulated automatic actuation of the system ~~throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position.~~ LA.2 Actual injection of coolant into the reactor vessel may be excluded from this test.

A.1

SURVEILLANCE REQUIREMENTS (Continued)

~~2. Performing a CHANNEL CALIBRATION of the:~~

~~a) Discharge line "keep filled" pressure alarm instrumentation and verifying the:~~

~~1) High pressure setpoint allowable value and the low pressure setpoint allowable value of the:~~

~~(a) LPCS system to be ≤500 psig and ≥45.5 psig, respectively.~~

~~(b) LPCI subsystem "A" to be ≤400 psig and ≥41.0 psig, respectively.~~

~~(c) LPCI subsystem "B" to be ≤400 psig and ≥38.5 psig, respectively.~~

~~(d) LPCI subsystem "C" to be ≤400 psig and ≥45.0 psig, respectively.~~

~~2) Low pressure setpoint allowable value of the HPCS system to be ≥42.5 psig.~~

~~b) Header delta P instrumentation and verifying the setpoint allowable value of the:~~

~~1) LPCS system and LPCI subsystems to be ± 1 psid.~~

~~2) HPCS system to be 5 ± 2.0 psid greater than the normal indicated ΔP.~~

L.5

3. Deleted

~~4. Visually inspecting the ECCS corner room watertight door seals and room penetration seals and verifying no abnormal degradation, damage, or obstructions.~~ LA.3

d. For the ADS by:

1. At least once per 31 days:

SR 3.5.1.3 a) Verify ADS accumulator supply header pressure is ≥ 150 psig.

SR 3.5.1.4 b) Verify ADS accumulator backup compressed gas system bottle pressure is ≥ 500 psig.

2. At least once per ~~18~~ 24 months* LD.1

SR 3.5.1.7 a) Performing a system functional test which includes an actual or simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation. L.4

SR 3.5.1.8 b) Manually opening each ADS valve ~~and observing the expected change in the indicated valve position.~~ LA.2

on a STAGGERED TEST BASIS for each valve solenoid (SR 3.5.1.8 only) M.1

ITS 3.5.2

A.1

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

ECCS injection/spray subsystems

LA.1

3.5.2 At least two ~~of the following~~ shall be OPERABLE:

a. ~~The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~

b. ~~Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

c. ~~Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

d. ~~Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

e. ~~The high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.~~

Add LCO Note L.4

APPLICABILITY: OPERATIONAL CONDITION 4 or 5*.

ACTION:

Action A
Action B

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

L.1

Action C
Action D

b. With both of the above required subsystems/systems inoperable, suspend ~~CORE ALTERATIONS and~~ all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish ~~SECONDARY~~ CONTAINMENT INTEGRITY within the next 8 hours.

A.2

A.3

APPL

*The ECCS is not required to be OPERABLE provided ~~that the reactor vessel head is removed, the cavity is flooded,~~ the spent fuel pool gates are removed, and water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

A.4

ITS 3.5.2

A.1

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3
SR 3.5.2.4
SR 3.5.2.5
SR 3.5.2.6

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1~~, except that the header delta P instrumentation is not required to be OPERABLE.~~

A.5

EMERGENCY CORE COOLING SYSTEMS A.1

3/4.5.3 SUPPRESSION CHAMBER~~#~~ A.8

LIMITING CONDITION FOR OPERATION

Moved to ITS 3.6.2.2 A.6

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft³, equivalent to a level of -4 1/2 inches.**

b. In OPERATIONAL CONDITION 4 or 5* with ~~a contained water volume of at least 70,000 ft³, equivalent to~~ a level of -12 feet 7 inches.~~**~~ LA.2 LA.3

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

Moved to ITS 3.6.2.2 A.6

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION C — b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit, suspend ~~CORE ALTERATIONS and~~ all operations that have a potential for draining the reactor vessel and ~~lock the reactor mode switch in the Shutdown position.~~ L.1

ACTION D — Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. A.3

L.2 A.2

ADD PROPOSED Required Action C.2 L.3

~~#See Specification 3.6.2.1 for pressure suppression requirements.~~ A.8

APPL — *The suppression chamber is not required to be OPERABLE ~~provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool,~~ the spent fuel pool gates are removed ~~when the cavity is flooded,~~ and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9. A.4 M.1

~~**Level is referenced to a plant elevation of 699 feet 11 inches (see Figure B 3/4.6.2-1).~~ LA.3

A.1

ITS 3.6.2.2

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 SUPPRESSION CHAMBER#

A.2

LIMITING CONDITION FOR OPERATION

LCO 3.6.2.2

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft^3, equivalent to a level of -4 1/2 inches.**

LA.1

b. In OPERATIONAL CONDITION 4 or 5* with a contained water volume of at least 70,000 ft^3, equivalent to a level of -12 feet 7 inches.**

A.3

Moved to ITS 3.5.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

2

L.1

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION A

ACTION B

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#See Specification 3.6.2.1 for pressure suppression requirements.

A.2

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

A.3

Moved to ITS 3.5.2

**Level is referenced to a plant elevation of 699 feet 11 inches (see Figure B 3/4.6.2-1).

LA.1

A.1

ITS 3.5.2

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

SR 3.5.2.1
SR 3.5.2.2

a. The water level to be greater than or equal to, as applicable:

1. -4 1/2 inches** at least once per 24 hours. — Moved to ITS 3.6.2.2 — A.6

2. -12 feet 7 inches~~**~~ at least once per 12 hours. — LA.3

~~4.5.3.2 With the suppression chamber level less than the above limit in OPERATIONAL CONDITION 5*, at least once per 12 hours verify footnote conditions* to be satisfied.~~ — A.7

~~*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.~~ — A.7

~~**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).~~ — LA.3

A.1

ITS 3.6.2.2

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

SR 3.6.2.2.1 a. The water level to be greater than or equal to, as applicable:

1. -4 1/2 inches~~**~~ at least once per 24 hours.

LA.1

2. -12 feet 7 inches** at least once per 12 hours.

4.5.3.2 With the suppression chamber level less than the above limit in OPERATIONAL CONDITION 5*, at least once per 12 hours verify footnote conditions* to be satisfied.

A.3

moved to ITS 3.5.2

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*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

A.3

moved to ITS 3.5.2

~~**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).~~

LA.1

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

A.1

LIMITING CONDITION FOR OPERATION

ITS 3.6.1.1

LCO 3.6.1.1 3.6.1.1 PRIMARY CONTAINMENT ~~INTEGRITY~~ shall be maintained. OPERABLE A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, ~~*~~ and 3. A3

ACTION:

A.2

ACTION A Without PRIMARY CONTAINMENT ~~INTEGRITY~~, restore PRIMARY CONTAINMENT ~~INTEGRITY~~ within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in

ACTION B COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT ~~INTEGRITY~~ shall be demonstrated: OPERABLE A.2

a. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.6.3. A.4 moved to ITS 3.6.1.3

SR 3.6.1.1.1 b. Perform required visual examinations and leakage rate testing except for primary containment air lock testing and main steam lines through the isolation valves, in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program.

~~*See Special Test Exception 3.10.1.~~ A3

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days. A.4 moved to ITS 3.6.1.3

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A.1

ITS 3.6.1.3

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2,* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

See ITS 3.6.1.1

L.10

add proposed Notes 1 and 2 to Required Actions A.2 and C.2 and SR 3.6.1.3.2 and SR 3.6.1.3.3

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

Required Actions A.2 and C.2 and SR 3.6.1.3.2

a. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and not locked, sealed, or secured and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.6.3. or check valves with flow secured

SR 3.6.1.3.2

Required Actions A.2 and C.2

L.11

L.2

Note 1 to Actions, Note 2 to SR 3.6.1.3.2, and Note 2 to SR 3.6.1.3.3

L.5

b. Perform required visual examinations and leakage rate testing except for primary containment air lock testing and main steam lines through the isolation valves, in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program.

See ITS 3.6.1.1

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*See Special Test Exception 3.10.1.

Required Action A.2 and SR 3.6.1.3.3

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are not locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

L.11

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CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

A.1

SURVEILLANCE REQUIREMENTS (Continued)

ITS 3.6.1.1

A.5

~~c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.~~

~~d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.~~

SR3.6.1.1.2 e. Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons. The frequency shall be in accordance with the Inservice Inspection Program for Post Tensioning Tendons.

ITS 3.6.1.2

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT AIR LOCKS A.1

LIMITING CONDITION FOR OPERATION

LCO 3.6.1.2 3.6.1.3 Each primary containment air lock shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2~~*~~, and 3. A.2

ACTION: add proposed ACTIONS Note 1 L.1

add proposed ACTIONS Note 2 A.3

Action A

a. With one primary containment air lock door inoperable: add proposed Note 1 to Required Action A A.4 add proposed Note 2 to Required Action A L.2

1. ~~Maintain~~ verify at least the OPERABLE air lock door closed within 1 hour and ~~either restore the inoperable air lock door to OPERABLE status~~ within 24 hours ~~or~~ lock the OPERABLE air lock door closed. A.5 L.3

2. ~~Operation may then continue until performance of the next required overall air lock leakage test provided~~ that the OPERABLE air lock door is verified to be locked closed at least once per 31 days. add proposed Note to Required Action A.3 A.6 L.4

Action D

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~4. The provisions of Specification 3.0.4 are not applicable.~~ A.6

Action C

b. With the primary containment air lock inoperable, or inoperable interlock mechanism except as a result of an inoperable air lock door, ~~maintain~~ verify at least one air lock door closed within 1 hour, restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. L.5 L.3

Action D

add proposed Required Action C.1 A.3

L.3

add proposed Action B L.5

~~*See Special Test Exception 3.10.1.~~ A.2

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ITS 3.6.1.2

A.1

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

SR 3.6.1.2.1 a. By performing required primary containment air lock leakage testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program,*.

SR 3.6.1.2.2 b. At least once per ~~6~~ 24 months by verifying that only one door in each air lock can be opened at a time.~~**~~ L.6

add proposed Note 1 to SR 3.6.1.2.1 A.3

Note 2 to SR 3.6.1.2.1 *Results shall be evaluated against acceptance criteria applicable to Specification 4.6.1.1.b.

~~**Only required to be performed upon entry into primary containment air lock when the primary containment is de-inerted.~~ L.6

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ITS 3.6.1.4

A.1

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

LCO 3.6.1.4

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between - 0.5 and +2.0 psig.

+0.75 M.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Action A: With the drywell and suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limits within 1 hour

Action B: or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits at least once per 12 hours.

A.1

ITS 3.6.1.5

CONTAINMENT SYSTEMS

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

LCO 3.6.1.5

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

ACTION A With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN
ACTION B within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1

4.6.1.7 The drywell average air temperature ~~shall be the average temperature of the operating return air plenum upstream of the primary containment ventilation heat exchanger coil and cabinet at the following locations and~~ shall be determined to be within the limit at least once per 24 hours:

LA.1

	~~Elevation~~	~~Azimuth~~
~~a.~~	~~740'0"~~	~~248°~~
~~b.~~	~~740'0"~~	~~76°~~

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

A.7

LCO 3.6.1.3

3.6.1.8 The drywell and suppression chamber purge system may be in operation with the drywell or suppression chamber purge supply and exhaust butterfly isolation valves open for inerting, deinerting, and pressure control. Purging through the Standby Gas Treatment System shall be restricted to less than or equal to 90 hours per 365 days.

L.12

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

add proposed ACTIONS Note 1 — L.5

ACTION:

add proposed ACTIONS Note 2 — A.2

ACTIONS A and C

With any drywell or suppression chamber purge supply or exhaust butterfly isolation valve open ~~for other than inerting, deinerting, or pressure control~~ close the butterfly valve(s) within 1 hour or

ACTION E

be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

L.1

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 The cumulative time that the drywell and suppression chamber purge system has been in operation purging through the Standby Gas Treatment System shall be verified to be less than or equal to 90 hours per 365 days prior to use in this mode of operation.

L.12

add proposed SR 3.6.1.3.1 — M.3

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

A.1

ITS 3.6.1.1

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a
2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:
   a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.
   b) 120°F with the main steam line isolation valves closed following a scram.

L.3

LCO 3.6.1.1

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

A.6

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.
2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

<See ITS 3.6.2.1 and ITS 3.6.2.2>

ITS 3.6.2.1

A.1

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

<See ITS 3.6.2.2>

LIMITING CONDITION FOR OPERATION

LCO 3.6.2.1

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

LCO 3.6.2.1.a 2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2 except that the maximum average temperature may be permitted to increase to: A.2

LCO 3.6.2.1.b a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

CONDITION D b) 120°F with the main steam line isolation valves closed following a scram. M.1

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2. A.3 moved to ITS 3.6.1.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION A b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in

ACTION B at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above: A.2

ACTION C 1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode. L.1

ACTION D 2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours. M.2 and be in MODE 4 in 36 hours

---

#See Specification 3.5.3 for ECCS requirements.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

ITS 3.6.2.2

A.1

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER~~#~~ A.2

LIMITING CONDITION FOR OPERATION

LCO 3.6.2.2

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. ~~Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a~~ LA.1 level between +3 inches~~**~~ and -4 1/2 inches~~**~~, and a

2. Maximum average temperature of 105°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

   b) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable A/$\sqrt{k}$ design value of 0.03 ft^2.

(See ITS 3.6.2.1)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

ACTION A a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within ~~1~~ 2 hour ~~or be in~~ L.1

ACTION B ~~at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN~~ within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 105°F, stop all testing which adds heat to the suppression pool, and restore the average temperature to less than or equal to 105°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

2. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

~~#See Specification 3.5.3 for ECCS requirements.~~ A.2

~~**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).~~ LA.1

A.1

ITS 3.6.1.1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. Deleted.

d. Deleted.

within one hour, or be Mode 3 in 12 hours, and Mode 4 in 36 hours.

L.1

ACTION A e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit ~~prior to increasing reactor coolant temperature above 200°F.~~

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 105°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per 60 minutes when suppression chamber average water temperature is greater than 105°F, by verifying suppression chamber average water temperature less than or equal to 110°F and THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 105°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

<See ITS 3.6.2.1 and ITS 3.6.2.2>

ITS 3.6.2.1

A.1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. Deleted.

d. Deleted.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

A.3 moved to ITS 3.6.1.1

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

see ITS 3.6.2.2

SR 3.6.2.1.1

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 105°F, except:

A.2

M.3

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

Required Action A.2

2. At least once per 60 minutes when suppression chamber average water temperature is greater than 105°F, by verifying suppression chamber average water temperature less than or equal to 110°F and THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

L.2

Required Action C.2

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 105°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

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ITS 3.6.2.2

A.1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. Deleted.

d. Deleted.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

<See ITS 3.6.2.1>

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

SR 3.6.2.2.1 a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 105°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per 60 minutes when suppression chamber average water temperature is greater than 105°F, by verifying suppression chamber average water temperature less than or equal to 110°F and THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 105°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

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A.1

ITS 3.6.1.1

SURVEILLANCE REQUIREMENTS (Continued)

c. Deleted. LD.1

SR 3.6.1.1.3 d. By conducting drywell-to-suppression chamber bypass leak tests at least once per (24) ~~18~~ months at an initial differential pressure of 1.5 psi and verifying that the A/√k calculated from the measured leakage is within the specified limit. LA.1 A.6

~~If any 1.5 psi leak test results in a calculated A/√k >20% of the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.~~ L.5

~~If two consecutive 1.5 psi leak tests result in a calculated A/√k greater than the specified limit, then:~~

~~1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated A/√k within the specified limits, and~~ L.4

~~2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.~~

~~If any required 5 psi leak test results in a calculated A/√k greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.~~ L.2

~~If two consecutive 5 psi leak tests result in a calculated A/√k greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.~~

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A.1

ITS 3.6.2.1

SURVEILLANCE REQUIREMENTS (Continued)

c. Deleted.

d. By conducting drywell-to-suppression chamber bypass leak tests at least once per 18 months at an initial differential pressure of 1.5 psi and verifying that the A/√k calculated from the measured leakage is within the specified limit.

If any 1.5 psi leak test results in a calculated A/√k >20% of the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 1.5 psi leak tests result in a calculated A/√k greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated A/√k within the specified limits, and
2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

If any required 5 psi leak test results in a calculated A/√k greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 5 psi leak tests result in a calculated A/√k greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

A.3
moved to
ITS 3.6.1.1

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A.1

ITS 3.6.2.4

CONTAINMENT SYSTEMS

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

LCO3.6.2.4

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of: LA.1

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Action A a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN
Action C within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Action B b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT
Action C SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours. A.2

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

SR3.6.2.4.1 a. At least once per 31 days by verifying that each valve (manual, power-operated, ~~or automatic~~), in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. A.3 or can be aligned to the correct position

SR3.6.2.4.2 b. By verifying that each of the required RHR pumps develops a flow of at least 450 gpm on recirculation flow through the suppression pool spray sparger when tested pursuant to Specification 4.0.5.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods. A.2

Page 2 of 2

ITS 3.6.2.3

A.1

CONTAINMENT SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

LCO 3.6.2.3

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of: LA.1

a. One OPERABLE RHR pump; and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHRSW heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

ACTION A a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within ~~12 hours~~ 7 days L.1 or be in at least
ACTION C HOT SHUTDOWN within the next 12 hours ~~and in COLD SHUTDOWN within the following 24 hours.~~

restore one subsystem to OPERABLE status within 8 hours

ACTION B b. With both suppression pool cooling loops inoperable, be in at least
ACTION C ~~HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN~~ within the next 24 hours. A.2

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

SR3.6.2.3.1 a. At least once per 31 days by verifying that each valve (manual, power-operated, ~~or automatic~~), in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. A.3

or can be aligned to the correct position

SR3.6.2.3.2 b. By verifying that each of the required RHR pumps develops a flow of at least 7200 gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

---

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods. A.2

Page 2 of 2

A.1

ITS 3.6.1.3

CONTAINMENT SYSTEMS

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.6.1.3 3.6.3 Each primary containment isolation valve and reactor instrumentation line excess flow check valve shall be OPERABLE**.

add proposed Applicability — M.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

add proposed Note 2 to ACTIONS — A.2

ACTION:

add proposed Notes 3 and 4 to ACTIONS — A.3

ACTIONS A and C a. With one or more of the primary containment isolation valves, except the reactor instrumentation line excess flow check valves, inoperable:

1. Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within ④ hours either; — A.4 — L.1

   a) Restore the inoperable valve(s) to OPERABLE status, or — A.5

   b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.* — or check valve with flow secured — L.2 Required Action A.1 only

ACTION E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. — L.3

add proposed ACTION B

ACTION C b. With one or more of the reactor instrumentation line excess flow check valves inoperable:

1. Operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within ④ hours ~~either~~: — 72 — A.6 — L.4

   a) The inoperable valve is returned to OPERABLE status, or — A.5

   b) The instrument line is isolated and the associated instrument is declared inoperable. — Note 3 to ACTIONS

ACTION E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

add proposed ACTION D — L.13

add proposed ACTION F — M.1

Note 1 to ACTIONS — *Isolation valves closed to satify these requirements may be reopened on an intermittent basis under administrative control. — L.5

Note 2 to SR 3.6.1.3.2 and SR 3.6.1.3.3 — **Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

A.1

ITS 3.6.1.3

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time. L.6 L.7

SR 3.6.1.3.7 4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 24 months by verifying that on a containment isolation test signal or actual each automatic isolation valve actuates to its isolation position. LD.1 L.8

SR 3.6.1.3.5 4.6.3.3 The isolation time of each primary containment power operated or automatic isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5. LA.1 LD.1

SR 3.6.1.3.8 4.6.3.4 Each reactor instrumentation line excess flow check valve shall be demonstrated OPERABLE at least once per 18 24 months by verifying that the valve checks flow. actuates to the isolation position L.9

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

SR 3.6.1.3.4 a. At least once per 31 days by verifying the continuity of the explosive charge. LD.1

SR 3.6.1.3.9 b. At least once per 18 24 months by removing the explosive squib from at least one explosive valve such that the explosive squib in each ~~explosive valve will be tested at least once per 90 months,~~ 120 and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No explosive squib shall remain in use beyond the expiration of its shelf-life and operating-life. definition of STAGGERED TEST BASIS LD.1 LA.2

4.6.3.6 At the frequency specified by the Primary Containment Leakage Rate Testing Program:

SR 3.6.1.3.10 a. Verify leakage rate for any one main steamline through the isolation valves is ≤ 100 scfh, not to exceed 400 scfh for all four main steamlines when tested at ≥ 25.0 psig.

SR 3.6.1.3.11 b. Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.

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ITS 3.6.1.6

A.1

CONTAINMENT SYSTEMS

3/4.6.4 VACUUM RELIEF

LIMITING CONDITION FOR OPERATION

LCO 3.6.1.6 3.6.4 All suppression chamber - drywell vacuum breakers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

ACTION A a. With one suppression chamber - drywell vacuum breaker inoperable for opening, restore the inoperable vacuum breaker to OPERABLE status within 72 hours or be

ACTION C in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION B b. With one suppression chamber -drywell vacuum breaker inoperable and open, within 4 hours close the manual isolation valves on both sides of the inoperable and open vacuum breaker. Restore the inoperable vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours

ACTION C and in COLD SHUTDOWN within the following 24 hours.

add proposed ACTION D — M.1

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

SR 3.6.1.6.1 a. Verified closed at least once per 14 days*.

b. Demonstrated OPERABLE: — L.1

SR 3.6.1.6.2 1. At least once per ~~91~~ 92 days and within 12 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

SR 3.6.1.6.3 2. At least once per ~~18~~ 24 months by verifying the force required to open the vacuum breaker from the closed position to be less than or equal to 0.5 psid. — LD.1 — LA.1

SR 3.6.1.6.1 Notes 1 and 2 *Surveillance Requirement 4.6.4.1.a is not required to be met for suppression chamber - drywell vacuum breakers that are open during Surveillances or for suppression chamber - drywell vacuum breakers that are functioning for pressure relief during normal and off-normal plant operations.

Page 2 of 2

ITS 3.6.4.1

A.1

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.6.4.1 3.6.5.1 SECONDARY CONTAINMENT ~~INTEGRITY~~ shall be ~~maintained~~. OPERABLE — A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

Without SECONDARY CONTAINMENT ~~INTEGRITY~~: OPERABLE

to OPERABLE status

Action A a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT ~~INTEGRITY~~ within 4 hours or be in at least HOT SHUTDOWN within the

Action B ~~next 12 hours~~ and in COLD SHUTDOWN within the following 24 hours.

Action C b. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

OPERABILITY — A.2

4.6.5.1 SECONDARY CONTAINMENT ~~INTEGRITY~~ shall be demonstrated by:

SR 3.6.4.1.1 a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inch of vacuum water gauge.# — M.1

b. Verifying at least once per 31 days that: — A.3

SR 3.6.4.1.2 1. At least one door in each access to the secondary containment is closed.

2. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position. — A.4 moved to ITS 3.6.4.2 — LD.1

c. At least once per ~~18~~ 24 months: on a STAGGERED TEST BASIS — M.2

SR 3.6.4.1.3 1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 300 seconds, and

SR 3.6.4.1.4 2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the secondary containment at a flow rate not exceeding 4000 cfm ± 10%.

Applicability *When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

~~#SECONDARY CONTAINMENT INTEGRITY is maintained when secondary containment vacuum is less than required for up to 1 hour solely due to Reactor Building ventilation system failure.~~ — M.1

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A.1

ITS 3.6.4.2

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inch of vacuum water gauge.#

See ITS 3.6.4.1

L.6

Required Action A.2 and SR 3.6.4.2.1

b. Verifying at least once per 31 days that:

1. At least one door in each access to the secondary containment is closed.

add proposed Required Action A.2 Note and SR 3.6.4.2.1 Note 1

SR 3.6.4.2.1

Required Action A.2

2. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

add proposed SR 3.6.4.2.1 Note 2

L.1

not locked, sealed, or otherwise secured

L.7

c. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 300 seconds, and

2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the secondary containment at a flow rate not exceeding 4000 cfm ± 10%.

---

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#SECONDARY CONTAINMENT INTEGRITY is maintained when secondary containment vacuum is less than required for up to 1 hour solely due to Reactor Building ventilation system failure.

See ITS 3.6.4.1

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A.1

ITS 3.6.4.2

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

LCO 3.6.4.2

3.6.5.2 The secondary containment ~~ventilation system automatic~~ isolation dampers ~~shown in Table 3.6.5.2-1~~ shall be OPERABLE ~~with isolation times equal to or less than shown in Table 3.6.5.2-1.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *. L.1

ACTION:

add proposed Note 1 to ACTIONS A.2

add proposed Notes 2 and 3 to ACTIONS

With one or more of the secondary containment ~~ventilation system automatic~~ isolation dampers ~~shown in Table 3.6.5.2-1~~ inoperable: LA.1

ACTION A

a. ~~Maintain at least one isolation damper OPERABLE in each affected penetration that is open~~ and within 8 hours, either: A.3

1. ~~Restore the inoperable damper to OPERABLE status, or~~ A.4
2. Isolate each affected penetration by use of at least one deactivated automatic damper secured in the isolation position, or
3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

add proposed ACTION B L.2

ACTION C

b. Otherwise, in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION D

c. Otherwise, in OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment ~~ventilation system automatic~~ isolation damper ~~shown in Table 3.6.5.2-1~~ shall be demonstrated OPERABLE: LA.1

a. ~~Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit by cycling the damper through at least one complete cycle of full travel and verifying the specified isolation time.~~ L.3

SR 3.6.4.2.3 b. During ~~COLD SHUTDOWN or REFUELING~~ at least once per ~~18~~ 24 months by verifying that on a containment isolation actual or test signal each isolation damper actuates to its isolation position. L.4 LD.1 L.5

SR 3.6.4.2.2 c. By verifying the isolation time to be within the limit when tested ~~pursuant to Specification 4.0.5.~~ every 92 days A.1

---

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

Applicability

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

DAMPER FUNCTION	ISOLATION TIME (Seconds)
1. Reactor Building Ventilation Supply Damper 2VR-04YA	10
2. Reactor Building Ventilation Supply Damper 2VR-04YB	10
3. Reactor Building Ventilation Exhaust Damper 2VR-05YA	10
4. Reactor Building Ventilation Exhaust Damper 2VR-05YB	10
5. Reactor Building Purge Train Isolation Damper 2VQ-037	90
6. Reactor Building Purge Train Isolation Damper 2VQ-038	90

LA.1

A.1

ITS 3.6.4.2

ITS 3.6.4.3

A.1

CONTAINMENT SYSTEMS

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.6.4.3 — LA.1

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.# — A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

Action A a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

Action B 1. In OPERABLE CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

add proposed Required Action C.1 — L.1

Action C 2. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

Action E b. With both standby gas treatment subsystems inoperable in OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

add proposed Action D — A.3

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

SR 3.6.4.3.1 a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters ~~OPERABLE~~. — LA.2

operating — A.4

Applicability

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#The normal or emergency power source may be inoperable in OPERATIONAL CONDITION *. — A.2

A.1

ITS 3.3.6.2

SURVEILLANCE REQUIREMENTS (Continued)

b. Perform required standby gas treatment filter testing in accordance with, and at the frequency specified by, the Ventilation Filter Testing Program.

c. Deleted.

24

LD.1

d. At least once per 18 months by:

1. Deleted.

A.8

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

   a. Reactor Building exhaust plenum radiation - high,

   b. Drywell pressure - high,

   c. Reactor vessel water level - low low, level 2, and

   d. Fuel pool vent exhaust radiation - high.

3. Deleted.

SR 3.3.6.2.4 for Functions 1, 2, 3 and 4

(See ITS 3.6.4.3)

Page 22 of 22

A.1

ITS 3.6.4.3

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.6.4.3.2 b. Perform required standby gas treatment filter testing in accordance with, and at the frequency specified by, the Ventilation Filter Testing Program.

c. Deleted.

24 LD.1

d. At least once per ~~18~~ months by:

1. Deleted.

SR 3.6.4.3.3 2. Verifying that the filter train starts and isolation dampers open on each of the following test signals: or actual L.2

A.5

a. Reactor Building exhaust plenum radiation - high,

b. Drywell pressure - high,

c. Reactor vessel water level - low low, level 2, and

d. Fuel pool vent exhaust radiation - high.

3. Deleted.

ITS 3.6.4.3

A.1

SURVEILLANCE REQUIREMENTS (Continued)

e. Deleted.

f. Deleted.

Page 6 of 6

A.1

ITS 3.6.3.1

CONTAINMENT SYSTEMS

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

LA.1

LCO 3.6.3.1 3.6.6.1 Two ~~independent~~ drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: add proposed Note to ACTION A — L.1

ACTION A With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or

ACTION C ~~be in at least HOT SHUTDOWN within the next 12 hours.~~

add proposed ACTION B — L.2

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. ~~At least once per 92 days by cycling each flow control valve and recirculation valve through at least one complete cycle of full travel.~~ — L.3

SR 3.6.3.1.1 b. At least once per ~~18~~ 24 months by verifying, during a recombiner system functional test: — LD.1

1. ~~That the heaters are OPERABLE by determining that the current in each phase differs by less than or equal to 5% from the other phases and is within 5% of the value observed in the original acceptance test, corrected for line voltage differences.~~

2. ~~That the reaction chamber gas temperature increases to 1200 ± 25°F within 2 hours.~~

LA.2

c. At least once per ~~18~~ 24 months by: — LD.1

1. ~~Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.~~ — L.4

SR 3.6.3.1.2 2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test ~~within 30 minutes following the above required functional~~ test. ~~The resistance to ground for any heater phase shall be greater than or equal to 100,000 ohms.~~ — LA.2

Page 2 of 2

ITS 3.6.3.2

A.1

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

LCO 3.6.3.2

3.6.6.2 The drywell and suppression chamber atomosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1~~*~~, during the time period: A.2

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER, preliminary to a scheduled reactor shutdown.

ACTION:

ACTION A With the oxygen concentration in the drywell and/or suppression chamber exceeding the limit, restore the oxygen concentration to within the limit within 24 hours
ACTION B or be in at least ~~STARTUP~~ ≤15% RTP within the next 8 hours. A.3

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.1

4.6.6.2 The oxygen concentration in the drywell and suppression chamber shall be verified to be within the limit ~~within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and~~ at least once per 7 days thereafter. A.4

~~*See Special Test Exception 3.10.5.~~ A.2

Page 2 of 2

ITS 3.7.1

A.1

3/4.7 PLANT SYSTEMS

3/4.7.1 CORE STANDBY COOLING SYSTEM-EQUIPMENT COOLING WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.1 3.7.1.1 Two ~~independent~~ residual heat removal service water (RHRSW) system subsystems shall be OPERABLE, ~~with each subsystem comprised of:~~ LA.1

~~a. Two OPERABLE RHRSW pumps, and~~

~~b. An OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring the water through the associated RHR heat exchanger.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, ~~4 and 5~~ LA.2

ACTION:

Action A — a. In OPERATIONAL CONDITION 1, 2 or 3: 7 days L.1

1. With one RHRSW subsystem inoperable, restore the inoperable subsystem to OPERABLE status within ~~72 hours~~ or be in at least

Action C — HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. L.2

restore one inoperable subsystem in 8 hours or

Action B — 2. With both RHRSW subsystems inoperable, be in at least HOT SHUTDOWN

Action C — within 12 hours and in COLD SHUTDOWN* within the next 24 hours. A.2

Notes to Required Actions A.1 and B.1

b. In OPERATIONAL CONDITION 3 ~~or 4~~ with the RHRSW subsystem(s)# inoperable which is associated with an RHR shutdown cooling mode loop(s) required OPERABLE by Specification 3.4.9.1 ~~or 3.4.9.2, as applicable~~ declare the associated RHR shutdown cooling mode loop(s) inoperable and take the ACTION required to Specification 3.4.9.1 ~~or 3.4.9.2, as applicable~~. LA.2

~~c. In OPERATIONAL CONDITION 5 with the RHRSW subsystem# cooling mode loop(s) inoperable which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2.~~ LA.2

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 4.7.1.1 Each residual heat removal service water system subsystem shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position, or can be aligned to the correct position A.3

~~*Whenever both RHRSW subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~ A.2

~~#Only one pump per subsystem need be OPERABLE if sufficient for decay heat removal.~~ LA.2

A.1

ITS 3.7.2

PLANT SYSTEMS

DIESEL GENERATOR COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.2

3.7.1.2 The Independent Unit 2 Division 1, 2 and 3 and the Unit 1 Division 2 diesel generator cooling water subsystems shall be OPERABLE with each subsystem comprised of:

LA.1

a. One OPERABLE diesel generator cooling water pump, and

b. An OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring cooling water to the associated diesel generator.

LA.2

APPLICABILITY: When the diesel generator is required to be OPERABLE.

MODES 1, 2, and 3

ACTION:

Add proposed ACTIONS Note — A.2

Action A

With one or more diesel generator cooling water subsystems inoperable, declare the associated diesel generator inoperable ~~and take the ACTION required by Specifications 3.8.1.1 or 3.8.1.2, as applicable~~.

M.1

A.3

LA.2

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each of the above required diesel generator cooling water subsystems shall be demonstrated OPERABLE:

SR 3.7.2.1

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SR 3.7.2.2

b. At least once per 18 months by verifying that:

24 — LD.1

1. Each pump starts automatically upon receipt of a start signal for the associated diesel generator, and

LA.3

actual or simulated — L.1

2. The Division 1 pump starts automatically upon receipt of a start signal for the LPCS pump to Unit 2

LA3

each required — LA3

actual or simulated — L.1

Page 2 of 2

A.1

ITS 3.7.3

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

LCO 3.7.3 3.7.1.3 The CSCS pond shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, ~~4, 5, and *~~. LA.1

ACTION A ACTION: With the CSCS pond inoperable due to sediment deposition in excess of limit or pond bottom elevation greater than limit, restore the pond to OPERABLE status within 90 days or: M.1

ACTION B a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Add 2nd portion of Condition B M.1

~~b. In OPERATIONAL CONDITION 4, 5, or *, declare the RHRSW system and the diesel generator cooling water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.~~ LA.1

SURVEILLANCE REQUIREMENTS

4.7.1.3 The CSCS pond shall be determined OPERABLE at least once per ~~18~~ 24 months by determining that: LD.1

SR 3.7.3.2 a. No sediment deposition in excess of 1.5 foot has occurred in the intake flume or in the CSCS pond ~~as determined by a series of sounding cross-sections compared to as-built soundings~~. LA.2

SR 3.7.3.3 b. The pond bottom elevation is less than or equal to 686.5 feet.

Add proposed SR 3.7.3.1 M.1

~~*When handling irradiated fuel in the secondary containment.~~ LA.1

A.1 ITS 3.7.4

3/4.7.2 CONTROL ROOM AND AUXILIARY ELECTRIC EQUIPMENT ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.4 3.7.2 Two independent control room and auxiliary electric equipment room emergency filtration system trains shall be OPERABLE#. LA.1 A.2

APPLICABILITY: ~~All OPERATIONAL CONDITIONS~~ and *. MODES 1, 2, and 3 During CORE ALTERATIONS During OPDRVs L.1

ACTION:

ACTION A a. With one emergency filtration system train inoperable, restore the inoperable train to OPERABLE status within 7 days or:

ACTION B 1. In OPERATIONAL CONDITIONS 1, 2, 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. L.1

ACTION C 2. ~~In OPERATIONAL CONDITION 4, 5~~ or *, initiate and maintain operation of the OPERABLE emergency filtration system in the pressurization mode of operation. During CORE ALTERATIONS, OPDRVs, Add proposed Required Actions C.2.1, C.2.2 + C.2.3 Add proposed ACTION D A.3

ACTION E b. With both emergency filtration system trains inoperable, ~~In OPERATIONAL CONDITION 4, 5~~ or *, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and initiate action to suspend operations with a potential for draining the reactor vessel. A.4

NOTE TO ACTION E c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room and auxiliary electric equipment room emergency filtration system train shall be demonstrated OPERABLE:

a. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ L.2

SR 3.7.4.1 1. Operate each Control Room and Auxiliary Electric Equipment Room Emergency Filter System for greater than or equal to 10 continuous hours with the heaters operating, and

SR 3.7.4.2 2. Manually initiating flow through the control room and auxiliary electric equipment room recirculation filters for at least 10 hours.

Applicability *When irradiated fuel is being handled in the secondary containment.

~~#The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4, 5 or *.~~ A.2

A.1

ITS 3.3.7.1

SURVEILLANCE REQUIREMENTS (Continued)

b. Perform required control room and auxiliary electric equipment room filter testing in accordance with, and at the frequency specified by, the Ventilation Filter Testing Program.

c. Deleted.

SR 3.3.7.1.4 d. At least once per ~~18~~ (24) months by: LD.1

1. Deleted.

⟨see ITS 3.7.4⟩

(Applicability change) — L.1

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A.1

ITS 3.7.4

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.4.3 b. Perform required control room and auxiliary electric equipment room filter testing in accordance with, and at the frequency specified by, the Ventilation Filter Testing Program.

c. Deleted. (24) LD.1

SR 3.7.4.4 d. At least once per ~~18~~ months by:

SR 3.7.4.5 1. Deleted.

page 5 of 6

A.1

SR 3.3.7.1.4

A.6

2. Verifying that on each of the below pressurization mode actuation test signals, the emergency train automatically switches to the pressurization mode of operation. Manually initiate flow through the control room and auxiliary electric equipment room recirculation filters and then verify that the control room and auxiliary electric equipment rooms are maintained at a positive pressure of greater than or equal to 1/8 inch W.G. relative to the adjacent areas during emergency train operation at a flow rate less than or equal to 4000 cfm:

   a) Outside air smoke detection, and

   b) Air intake radiation monitors.

(See ITS 3.7.4)

3. Deleted.

e. Deleted.

f. Deleted.

Page 11 of 11

A.1

ITS 3.7.4

SURVEILLANCE REQUIREMENTS (Continued)

L.3 actual or

LA.2

LA.4

SR 3.7.4.4 — 2. Verifying that on each of the below pressurization mode actuation test signals, the emergency train ~~automatically switches to the pressurization mode of operation.~~ Manually initiate flow through the control room and auxiliary electric equipment room recirculation filters and then verify that the control room and

LA.3

SR 3.7.4.5 — auxiliary electric equipment rooms are maintained at a positive pressure of greater than or equal to 1/8 inch W.G. relative to the adjacent areas during emergency train operation at a flow rate less than or equal to 4000 cfm:

actuates

LA.4

a) Outside air smoke detection, and — LA.2

b) Air intake radiation monitors. — LA.4

3. Deleted.

e. Deleted.

f. Deleted.

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PLANT SYSTEMS A.1

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.5.3 3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE ~~with an OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

~~a. With a RCIC discharge line "keep filled" pressure alarm instrumentation channnel inoperable, perform Surveillance Requirement 4.7.3.a.1 at least once per 24 hours.~~ L.2

ACTION A — b. With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12

ACTION B — hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

SR 3.5.3.1 — 1. Verifying ~~by venting at the high point vents~~ LA.2 that the system piping from the pump discharge valve to the system isolation valve is filled with water,

~~2. Performance of a CHANNEL FUNCTIONAL TEST of the discharge line "keep filled" pressure alarm instrumentation, and~~ L.2

SR 3.5.3.2 — 3. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

~~4. Verifying that the pump flow controller is in the correct position.~~ LA.2

SR 3.5.3.3 — b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

SR 3.5.3.3 NOTE — *The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow is adequate to perform the tests. A.2

PLANT SYSTEMS

A.1

SURVEILLANCE REQUIREMENTS

c. At least once per ~~18~~ months by: 24 — LD.1   actual or — L.1

SR 3.5.3.5 — 1. Performing a system functional test which includes actual or simulated automatic actuation ~~and verifying that each automatic valve in the flow path actuates to its correct position.~~ — LA.2 but may exclude

SR 3.5.3.5 NOTE — actual injection of coolant into the reactor vessel.

SR 3.5.3.4 — 2. Verifying that the system is capable of providing a flow of greater than or equal to 600 gpm to the reactor vessel when with a system head corresponding to reactor pressure — A.4 steam is supplied to the turbine at a pressure of 150 ± 15 psig using the test flow path.* — LA.3

~~3. Performing a CHANNEL CALIBRATION of the discharge line "keep filled" pressure alarm instrumentation and verifying the low pressure setpoint allowable value to be ≥29.0 psig.~~ — L.2

d. By demonstrating MCC-221y and the 250-volt battery and charger OPERABLE:

1. At least once per 7 days by verifying that:
   - a) MCC-221y is energized, and has correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 250 volts.
   - b) The electrolyte level of each pilot cell is above the plates,
   - c) The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.200, and
   - d) The overall battery voltage is greater than or equal to 250 volts.
2. At least once per 92 days by verifying that:
   - a) The voltage of each connected battery is greater than or equal to 250 volts under float charge and has not decreased more than 12 volts from the value observed during the original test,
   - b) The specific gravity, corrected to 77°F, of each connected cell is greater than or equal to 1.195 and has not decreased more than 0.05 from the value observed during the previous test, and
   - c) The electrolyte level of each connected cell is above the plates.
3. At least once per 18 months by verifying that:
   - a) The battery shows no visual indication of physical damage or abnormal deterioration, and
   - b) Battery terminal connections are clean, tight, free of corrosion and coated with anticorrosion material.

Moved to ITS 3.8.4, 3.8.6, and 3.8.7

A.3

SR 3.5.3.4 NOTE — *The provisions of Specification 4.0.4 are not applicably provided the surveillance is performed within 12 hours after reactor steam pressure and flow — A.2 is adequate to perform the tests.

add proposed LCO 3.8.4 and Applicability M.1

PLANT SYSTEMS A.1

SURVEILLANCE REQUIREMENTS

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.
2. Verifying that the system is capable of providing a flow of greater than or equal to 600 gpm to the reactor vessel when steam is supplied to the turbine at a pressure of 150 ± 15 psig using the test flow path.
3. Performing a CHANNEL CALIBRATION of the discharge line "keep filled" pressure alarm instrumentation and verifying the low pressure setpoint allowable value to be ≥29.0 psig.

Add proposed ITS LCO 3.8.4 (Division 1 250V requirements) and proposed ACTION C

d. By demonstrating MCC-221y and the 250-volt battery and charger OPERABLE: A6

SR 3.8.4.1

1. At least once per 7 days by verifying that:

   a) MCC-221y is energized, and has correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 250 volts.

   See ITS 3.5.3

   b) The electrolyte level of each pilot cell is above the plates,
   c) The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.200, and

   See ITS 3.5.3

   d) The overall battery voltage is greater than or equal to ~~250~~ 256 volts. on float charge A.7

2. At least once per 92 days by verifying that:

   a) The voltage of each connected battery is greater than or equal to 250 volts under float charge and has not decreased more than 12 volts from the value observed during the original test,
   b) The specific gravity, corrected to 77°F, of each connected cell is greater than or equal to 1.195 and has not decreased more than 0.05 from the value observed during the previous test, and
   c) The electrolyte level of each connected cell is above the plates.

M.3

3. At least once per 18 months by verifying that:

   SR 3.8.4.3 a) The battery shows no visual indication of physical damage or abnormal deterioration, and

   SR 3.8.4.4 b) Battery terminal connections are clean, tight, free of corrosion and coated with anticorrosion material. L.2

The provisions of Specification 4.0.4 are not applicably provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

<See ITS 3.5.3>

add proposed SR 3.8.4.2, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8 M.2

A.1

ITS 3.8.6

PLANT SYSTEMS ← add proposed LCO 3.8.6 and Applicability — A.6

SURVEILLANCE REQUIREMENTS ← add proposed ACTIONS A and B and ACTIONS Note — L.5

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.
2. Verifying that the system is capable of providing a flow of greater than or equal to 600 gpm to the reactor vessel when steam is supplied to the turbine at a pressure of 150 ± 15 psig using the test flow path.
3. Performing a CHANNEL CALIBRATION of the discharge line "keep filled" pressure alarm instrumentation and verifying the low pressure setpoint allowable value to be ≥29.0 psig.

<See ITS 3.5.3>

M.5 — add proposed Table 3.8.6-1 Cat A, Cat B, Cat C, second part, limits and footnote (a)

d. By demonstrating MCC-221y and the 250-volt battery and charger OPERABLE:

SR 3.8.6.1 1. At least once per 7 days by verifying that: Category A limits are met — M.4

a) MCC-221y is energized, and has correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 250 volts. <See ITS 3.8.7>

Table 3.8.6-1 Cat C limit { b) The electrolyte level of each pilot cell is above the plates. — L.3

Table 3.8.6-1 Cat A limit { c) The pilot cell specific gravity, ~~corrected to 77°F~~, is greater than or equal to 1.200, and — M.6

Add proposed Table 3.8.6-1 footnote (c)

d) The overall battery voltage is greater than or equal to 250 volts. <See ITS 3.8.4>

SR 3.8.6.2 2. At least once per 92 days by verifying that: Category B limits are met — M.4

Table 3.8.6-1 Cat B limit {

a) The voltage 2.13 of each connected ~~battery~~ cell is greater than or equal to 250 volts under float charge ~~and has not decreased more than 12 volts from the value observed during the original test~~, — L.6, L.6, M.6

b) The specific gravity, ~~corrected to 77°F~~, of each connected cell is greater than or equal to 1.195 ~~and has not decreased more than 0.05 from the value observed during the previous test~~, and ← Add proposed average limit — M.5, L.6

Table 3.8.6-1 Cat C limit { c) The electrolyte level of each connected cell is above the plates. add proposed Table 3.8.6-1 Cat A, Cat B, and Cat C, second part, limits and footnote (a) — M.5

Add proposed Table 3.8.6-1 footnote (c)

3. At least once per 18 months by verifying that:

a) The battery shows no visual indication of physical damage or abnormal deterioration, and
b) Battery terminal connections are clean, tight, free of corrosion and coated with anticorrosion material.

— L.3

← add proposed SR 3.8.6.3 — M.4

The provisions of Specification 4.0.4 are not applicably provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

<See ITS 3.8.4>

add proposed Table 3.8.6-1 Cat A and Cat C float voltage limits — L.6

add proposed Table 3.8.6-1 Cat C specific gravity limits, including footnotes (b) and (c) — M.5

A.1

ITS 3.8.7

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

add proposed ITS 3.8.7 Applicability

M.3

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.
2. Verifying that the system is capable of providing a flow of greater than or equal to 600 gpm to the reactor vessel when steam is supplied to the turbine at a pressure of 150 ± 15 psig using the test flow path.*
3. Performing a CHANNEL CALIBRATION of the discharge line "keep filled" pressure alarm instrumentation and verifying the low pressure setpoint allowable value to be >29.0 psig.

<See ITS 3.5.3>

d. By demonstrating MCC-221y and the 250-volt battery and charger OPERABLE:

<See ITS 3.5.3>

add proposed LCO 3.8.7 for Div 1 250V and ACTION F

A.4

1. At least once per 7 days by verifying that:

SR3.8.7.1

a) MCC-221y is energized and has correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 250 volts.

LA.1

LA.2

b) The electrolyte level of each pilot cell is above the plates.

c) The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.200, and

<See ITS 3.8.6>

d) The overall battery voltage is greater than or equal to 250 volts.

<See ITS 3.8.4>

2. At least once per 92 days by verifying that:

a) The voltage of each connected battery is greater than or equal to 250 volts under float charge and has not decreased more than 12 volts from the value observed during the original test,
b) The specific gravity, corrected to 77°F, of each connected cell is greater than or equal to 1.195 and has not decreased more than 0.05 from the value observed during the previous test, and
c) The electrolyte level of each connected cell is above the plates.

<See ITS 3.8.6>

3. At least once per 18 months by verifying that:

a) The battery shows no visual indication of physical damage or abnormal deterioration, and
b) Battery terminal connections are clean, tight, free of corrosion and coated with anticorrosion material.

<See ITS 3.8.4>

<See ITS 3.5.3>

---

*The provisions of Specification 4.0.4 are not applicably provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

CTS 3/4.7.4

3/4.7.4 SEALED SOURCE CONTAMINATION

R.1

LIMITING CONDITION FOR OPERATION

3.7.4 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

   a. The licensee, or

   b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.4.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

   a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

      1. With a half-life greater than 30 days, excluding Hydrogen 3, and

      2. In any form other than gas.

CTS 3/4.7.4

R.1

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source ~~and fission detector~~ shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.4.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

CTS 3/4.7.7

R,1

~~PLANT SYSTEMS~~

~~3/4.7.7 AREA TEMPERATURE MONITORING~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.7 The temperature of each area of Unit 1 and Unit 2 shown in Table 3.7.7-1 shall be maintained within the limits indicated in Table 3.7.7-1.~~

~~APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.~~

~~ACTION:~~

~~With one or more areas exceeding the temperature limit(s) shown in Table 3.7.7-1:~~

~~a. For more than 8 hours, in lieu of any Licensee Event Report, prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.~~

~~b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.7 The temperature in each of the above required areas shown in Table 3.7.7-1 shall be determined to be within its limit at least once per 24 hours.~~

page 3 of 4

CTS 3/4.7.7

R.1

TABLE 3.7.7-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
A. Unit 2 Area Temperature Monitoring	
1. Control Room	50-104
2. Auxiliary Electric Equipment Room	50-104
3. Diesel Generator Room	50-122
4. Switchgear Room	50-104
5. HPCS, LPCS, RHR & RCIC Rooms	50-150
6. Primary Containment	
a. Drywell	50-150
b. Beneath Reactor Pressure Vessel	50-185
B. Unit 1 Area Temperature Monitoring Required For Unit 2	
1. Auxiliary Electric Equipment Room	50-104
2. Diesel Generator 1A Room	50-122
3. Division 1 and 2 Switchgear Rooms	50-104

page 4 of 4

CTS 3/4.7.8

~~3/4.7.8 STRUCTURAL INTEGRITY OF CLASS 1 STRUCTURES~~

R.1

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.8 The structural integrity of Class 1 structures shall be verified pursuant to the requirements of Specifications 4.7.8.1 and 4.7.8.2.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~With the settlement of any Class 1 structure not verified to be within the allowable final settlement value as required, submit a Special Report in accordance with Specification 6.6.C:~~

a. ~~By telephone within 24 hours,~~

b. ~~Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and~~

c. ~~In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.8.1 The total settlement of each Class 1 structure and the differential settlement between Class 1 structures shall be determined to the nearest 0.01 foot by measurement and calculation:~~

a. ~~At least once per 31 days:~~

    1. ~~Until observed settlement has stabilized,* and~~

    2. ~~Whenever previously stabilized* settlement exceeds 0.01 foot since the previous reading.~~

b. ~~At least once per 6 months.~~

~~4.7.8.2 A Special Report shall be prepared and submitted to the Commission at least once per 6 months until settlement of Class 1 structures has stabilized. The report shall include settlement and differential settlement plots versus time and a comparison of allowable and actual settlement.~~

~~*≤ 0.01 foot from previous reading.~~

page 2 of 2

CTS 3/4.7.9

LA.1

~~3/4.7.9 SNUBBERS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE.~~

~~APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.~~

~~ACTION:~~

~~With one or more snubbers inoperable, on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.~~

~~a. Inspection Types~~

~~As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.~~

~~b. Visual Inspections~~

~~Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7.9-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7.9-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment 75.~~

~~c. Visual Inspection Acceptance Criteria~~

~~Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and~~

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LA.1

CTS 3/4.7.9

SURVEILLANCE REQUIREMENTS (Continued)

(3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.9f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspection

An inspection shall be performed of all hydralic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

At least once per 18 months during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested, in accordance with Figure 4.7-1. "C" is the

CTS 3/4.7.9

LA.1

e. Functional Tests (Continued)

total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type may be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

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CTS 3/4.7.9

LA.1

SURVEILLANCE REQUIREMENTS (Continued)

f. **Functional Test Acceptance Criteria**

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. **Functional Test Failure Analysis**

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

h. **Functional Testing of Repaired and Replaced Snubbers**

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test

CTS 3/4.7.9

LA.1

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~h. Functional Testing of Repaired and Replaced Snubbers (Continued)~~

~~criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.~~

~~i. Snubber Service Life Program~~

~~The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.5B.~~

CTS 3/4.7.9

LA.1

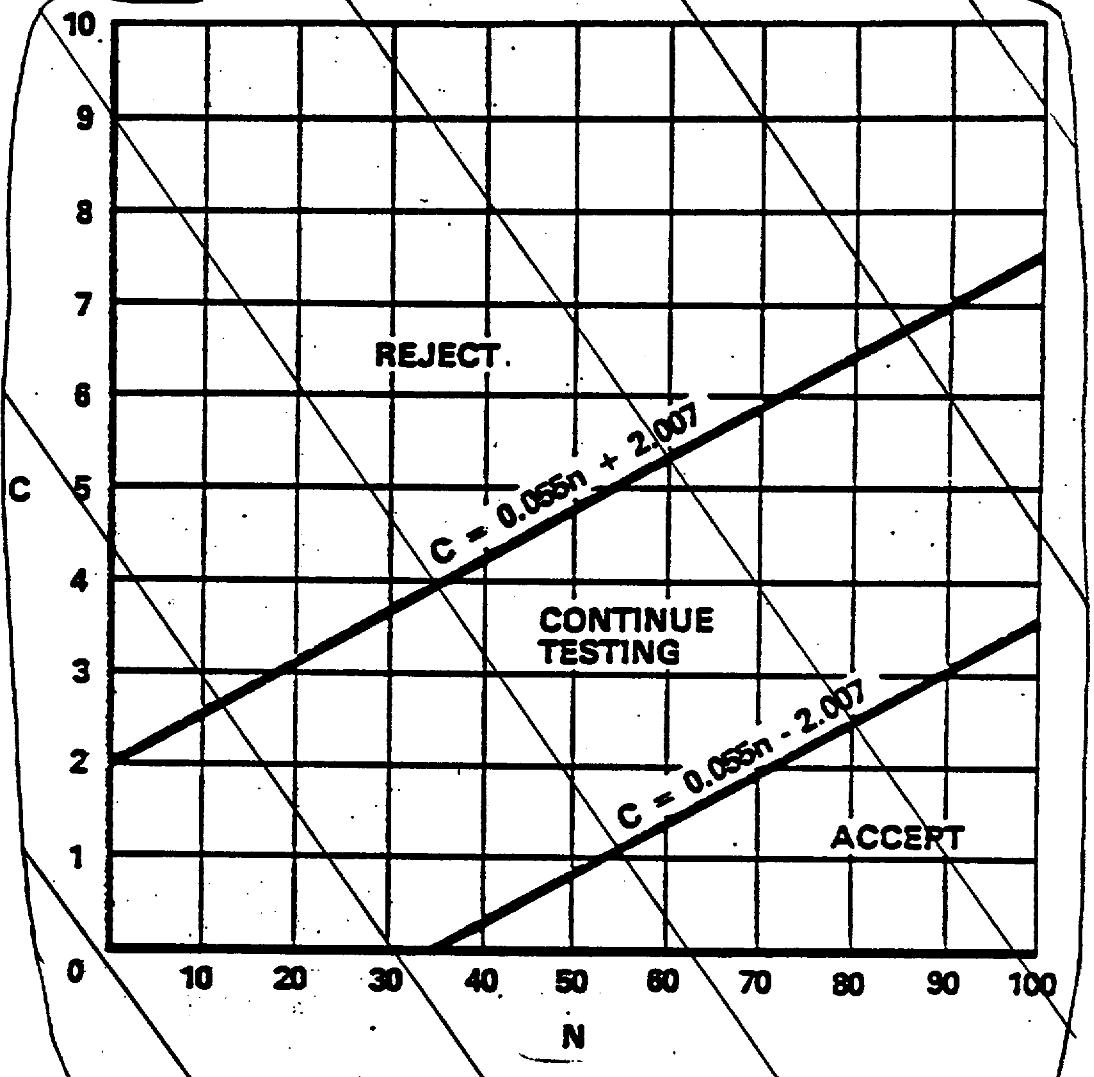

FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

CTS 3/4.7.9

LA,1

TABLE 4.7.9-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use the next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

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~~TABLE 4.7.9-1~~
~~SNUBBER VISUAL INSPECTION INTERVAL~~
~~(Continued)~~

~~Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.~~

~~Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.~~

~~Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.~~

~~Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.~~

LA.1

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A.1

ITS 3.7.7

3/4.7.10 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.7 3.7.10 The main turbine bypass system shall be OPERABLE. Add proposed 2nd part of LCO 3.7.7 A.2

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. A.3

ACTION:

With the main turbine bypass system inoperable:

ACTION A

1. If at least four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:
   - a) Within 2 hours, either:
     - 1) Restore the system to OPERABLE status, or
     - LCO 3.7.7 2) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) to the main turbine bypass inoperable value per Specification 3.2.3.
   - ACTION B b) Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

ACTION A

2. If less than four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:
   - a) Within 2 hours restore the system to OPERABLE status or A.4 increase the MCPR LCO to the main turbine LCO 3.7.7 bypass inoperable value per Specification 3.2.3, and
   - ~~b) Within the next 12 hours restore the system to OPERABLE status.~~ L.1
   - ACTION B c) Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

SR 3.7.7.1 a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel.

b. ~~18~~ 24 months by: LD.1

SR 3.7.7.2 1. Performing a system functional test ~~which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position~~. LA.1

SR 3.7.7.3 2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME ~~to be less than or equal to 200 milliseconds~~ is within limits LA.2

page 3 of 3

A.1

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.8.1 3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

qualified — LA.1

LCO 3.8.1.a a. Two ~~physically independent~~ circuits between the offsite transmission network and the onsite Class 1E distribution system, and — LA.1

LCO 3.8.1.b b. ~~Separate and independent~~ diesel generators* 0, ~~1A~~, 2A ~~and~~ 2B with:

LCO 3.8.1.c

SR 3.8.1.4 1. For diesel generator 0, 1A and 2A: — A.2

a) A separate day fuel tank containing a minimum of 250 gallons of fuel.

b) A separate fuel storage system containing a minimum of 31,000 gallons of fuel. — A.3 moved to ITS 3.8.3

SR 3.8.1.4 2. For diesel generator 2B, a separate fuel storage tank and a day tank containing a minimum of ~~29,750 gallons of fuel.~~ — A.2

550 gallons of fuel — M.2

3. ~~A separate fuel transfer pump.~~ — LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. — A.4 L.19

ACTION: Add proposed Applicability Notes

7 days — L.18

ACTION A a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within ~~72 hours~~ or be in at least ~~HOT SHUTDOWN within the next 12 hours and in COLD~~ SHUTDOWN within the following 24 hours.

ACTION G — Add proposed Required Action A.3 2nd Completion Time — L.1

ACTION C b. With either the 0 or 2A diesel generator inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE

~~*See page 3/4 8-1(a).~~ — A.5

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

A.1

ITS 3.8.3

LIMITING CONDITION FOR OPERATION

add proposed fuel oil storage tank and starting air LCO

A.2

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Separate and independent diesel generators* 0, 1A, 2A and 2B with:

1. For diesel generator 0, 1A and 2A:

   a) A separate day fuel tank containing a minimum of 250 gallons of fuel.

   b) A separate fuel storage system containing a minimum of 31,000 gallons of fuel.

SR 3.8.3.1

A.3

2. For diesel generator 2B, a separate fuel storage tank and a day tank containing a minimum of 29,750 gallons of fuel.

3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

A.2

ACTION:

a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With either the 0 or 2A diesel generator inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE

add proposed ACTIONS A, B, C, D, and E and ACTIONS Note

L.1

*See page 3/4 8-1(a).

<See ITS 3.8.1>

ITS 3.8.1

A.1

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

Condition B

*For the purposes of completing maintenance, modification, and/or technical specification surveillance requirements, on the ~~0~~ Division 1 diesel generator and its support systems during a refuel outage, as part of pre-planned maintenance, modifications, and/or the surveillance program, the requirements of action statement b are modified to:

1. ~~Eliminate the requirement for performing technical specification surveillance requirements 4.8.1.1.1.a on each operable AC source, immediately and once per 8 hours thereafter, when the 0 diesel generator is declared inoperable.~~ A.5

Required Action B.4 Condition C

2. Allow an additional 96 hours in excess of the 72 hours allowed in action statement b for the 0 diesel generator to be inoperable. L.1

Add proposed Required Action B.4 2nd Completion Time

Provided that the following conditions are met:

Note to Condition B

A. Unit 1 is in operational condition 4 or 5 or defueled prior to taking the 0 diesel generator out of service.

Required Action B.2 1st Completion Time

B. Surveillance requirements 4.8.1.1.1a ~~and 4.8.1.1.2a.4~~ are successfully completed, for the offsite power sources ~~and the 1A and 2A diesel generators,~~ within ~~48 hours prior to removal of the 0 diesel generator from service.~~ 1 hour L.2 M.4

C. ~~No maintenance is performed on the offsite circuits or the 1A or 2A diesel generators, while the 0 diesel generator is inoperable.~~ L.2

Required Action B.2 2nd Completion Time

D. Technical specification requirement 4.8.1.1.1a is performed daily, while the 0 diesel generator is inoperable.

Required Action B.1

E. ~~The control circuit for the unit cross-tie circuit breakers between buses 142Y and 242Y are temporarily modified to allow the breakers to be closed with a diesel generator feeding the bus, while the 0 diesel generator is inoperable.~~

Verify the unit crosstie breakers between the unit and opposite unit Division 2 4.16 kV emergency buses are capable of being closed with a DG powering one of the buses. LA.8

~~The provisions of technical specification 3.0.4 are not applicable.~~ M.5

ITS 3.8.1

A.1

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

Add proposed Required Action C.4 2nd Completion Time L.1 L.3

ACTION C
diesel generators, separately, by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours*, unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated. Restore the diesel generator to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
Action G

Add proposed Action E Note A.7

c. Condition E
With one offsite circuit of the above required A.C. sources and diesel generator 0 or 2A of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining
Required Actions A.1 and C.1
A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. (24) L.4
Required Action C.3.2
If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators, separately, by performing L.3 Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours*, unless the
Required Action C.3.1
absence of any potential common mode failure for the remaining diesel generator is demonstrated.
Required Actions E.1 and E.2
Restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in
Action G
at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. L.1
Required Actions A.3 and C.4
Restore at least two offsite circuits and diesel generators 0 and 2A to OPERABLE status within 12 ~~hours from the time of initial loss~~ or be in at least HOT SHUTDOWN
Action G
within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Add proposed Required Actions A.3 3rd Completion Time and C.4 2nd Completion Time

d. ACTION C
With diesel generator 2B of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the offsite A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators, separately, by performing L.3 Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours*, unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated. Restore diesel generator 2B to OPERABLE status within 72 hours or declare the HPCS system
Applicability Note 1
inoperable ~~and take the ACTION required by specification 3.5.1~~. A8

L.3
~~*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY. The provisions of Specification 3.0.2 are not applicable.~~

Add proposed Required Action C.4 2nd Completion Time L.1

A.1

M.6

LIMITING CONDITION FOR OPERATION (Continued)

Add proposed Required Action G.2

ACTION (Continued)

ACTION D e. With both of the above required offsite circuits inoperable, restore at least one offsite circuit to OPERABLE status within 24 hours, or be in at least HOT SHUTDOWN within the next 12 hours. ACTION G
ACTION A With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within ~~72 hours from the time of initial loss~~ or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. ACTION G L.1

ACTION F f. With diesel generators 0 and 2A of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and Surveillance Requirement 4.8.1.1.2.a.4 for the 2B and 1A diesel generators, separately, within ~~8~~ 24 hours*. ACTIONS B and C L.4 L.3
ACTION F Restore at least one of the inoperable diesel generators 0 or 2A to OPERABLE status within 2 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. ACTION G
ACTIONS B and C Restore both diesel generators 0 and 2A to OPERABLE status ~~within 72 hours, from the time of initial loss~~ or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. ACTION G L.1

ACTION C g. With diesel generator 1A of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the 1A diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the ~~2A~~ required OPERABLE diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours*, unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated. M.7 L.3
Applicability Note 2 Restore the inoperable diesel generator 1A to OPERABLE status within 72 hours or declare ~~standby gas treatment system subsystem A, Unit 1 drywell and suppression chamber hydrogen recombiner system, and control room and auxiliary electric equipment room emergency filtration system train A~~ associated required equipment ~~inoperable, and take the ACTION required by specifications 3.6.5.3, 3.6.6.1, and 3.7.2. Continued performance of Surveillance Requirement 4.8.1.1.1.a is not required provided the above systems are declared inoperable and the action of their respective specifications is taken.~~ LA.3 A.8 A.4 LA.3

~~*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY. The provisions of Specification 3.0.2 are not applicable.~~ L.3

ITS 3.8.1

A.1

ACTION (Continued)

M.8

ACTION E h. With one offsite circuit of the above required A.C. electrical power sources and diesel generator 2B inoperable, ~~apply the requirements of ACTION a and d specified above.~~

Action F i. With either diesel generators 0 or 2A inoperable and diesel generator 2B inoperable, apply the requirements of ACTION b and d specified above.

A.9

Actions A and C j. With one offsite circuit of the above required A.C. electrical power sources and diesel generator 1A inoperable, apply the requirements of ACTION a and g specified above.

Action C k. With diesel generator 2B and diesel generator 1A inoperable, apply the requirements of ACTION d and g specified above.

Action C l. With diesel generator 0 and diesel generator 1A inoperable, apply the requirements of ACTION b and g specified above.

Add proposed ACTION H A.10

Add proposed 2nd Condition of ACTION F L.5

ITS 3.8.1

A.1

A.18

Add proposed Surveillance Table Notes 1 and 2

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

SR 3.8.1.1 a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and LD.1

SR 3.8.1.8 b. Demonstrated OPERABLE at least once per ~~18~~ 24 months ~~during shutdown~~ by manually transferring unit power supply from the normal circuit to the alternate circuit. L.6

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:*

a. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ by: L.7

SR 3.8.1.4 1. Verifying the fuel level in the day fuel tank. Moved to ITS 3.8.3

2. Verifying the fuel level in the fuel storage tank. A.3

SR 3.8.1.6 3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank. once per 92 days L8

Achieves a generator voltage of 2400 V & frequency ≥ 58.8 Hz

SR 3.8.1.2 SR 3.8.1.7 4. Verifying the diesel starts from ambient condition and accelerates ~~to~~ 900 rpm + ~~5%~~, -2% in less than or equal to 13 seconds**. ~~The~~ generator voltage and frequency shall be 4160 ±150 volts and 60 ~~± 3.0~~ ± 1.2 Hz ~~within 13 seconds** after the start signal~~. Achieves a steady state L.9 A.11

Add proposed SR 3.8.1.2 Note 3 and SR 3.8.1.7 Note 2

SR 3.8.1.3 & SR 3.8.1.3 Note 1 5. Verifying the diesel generator is synchronized, and then loaded to 2400 kW to 2600 kW*** in accordance with the manufacturer's recommendations, and operates with this load for at least 60 minutes.

Add proposed SR 3.8.1.3 Notes 3 and 4 M.9

Add proposed SR 3.8.1.3 Note 5 A.11

SR 3.8.1.2 Note 4 SR 3.8.1.7 Note 1 *All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period ~~as recommended by the manufacturer~~. LA.4

SR 3.8.1.7 Frequency **Surveillance testing to verify the diesel generator start (13 second) time from ambient conditions shall be performed at least once per 184 days. SR 3.8.1.2 Note 1 All other engine starts performed for the purpose of meeting these surveillance requirements may be conducted in accordance with warmup and loading procedures, as recommended by the manufacturer, ~~in order to minimize mechanical stress and wear on the diesel generator caused by fast starting of the diesel generator~~. LA.4

SR 3.8.1.3 Note 2 ***Transients, outside of this load band, do not invalidate the surveillance tests.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by manually transferring unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:*

L.2

a. At least once per 31 days on a STAGGERED TEST BASIS by:

SR3.8.3.1 (Div3 DG, only) 1. Verifying the fuel level in the day fuel tank.

SR3.8.3.1 2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank.

4. Verifying the diesel starts from ambient condition and accelerates to 900 rpm + 5%, -2% in less than or equal to 13 seconds**. The generator voltage and frequency shall be 4160 ±150 volts and 60 + 3.0, -1.2 Hz within 13 seconds** after the start signal.

5. Verifying the diesel generator is synchronized, and then loaded to 2400 kW to 2600 kW*** in accordance with the manufacturer's recommendations, and operates with this load for at least 60 minutes.

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*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.

**Surveillance testing to verify the diesel generator start (13 second) time from ambient conditions shall be performed at least once per 184 days. All other engine starts performed for the purpose of meeting these surveillance requirements may be conducted in accordance with warmup and loading procedures, as recommended by the manufacturer, in order to minimize mechanical stress and wear on the diesel generator caused by fast starting of the diesel generator.

***Transients, outside of this load band, do not invalidate the surveillance tests.

(See ITS3.8.1)

ITS 3.8.1

A.1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

~~6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.~~ L.10

Moved to ITS 3.8.3 7. Verifying the pressure in required diesel generator air start receivers to be greater than or equal to 200 psig. A.3

SR 3.8.1.5 b. At least once per 31 days ~~and after each operation of the diesel where the period of operation was greater than or equal to 1 hour~~ by checking for and removing accumulated water from the day fuel tanks. L.11 A.3

Moved to ITS Section 5.5 c. By sampling and analyzing stored and new fuel oil in accordance with the following:

1. At least once per 92 days, and for new fuel oil prior to addition to the storage tanks, that a sample obtained and tested in accordance with the applicable ASTM Standards has:

   a) A water and sediment content within applicable ASTM limits.

   b) A kinematic viscosity at 40°C within applicable ASTM limits.

2. At least every 31 days, and for new fuel oil prior to addition to the storage tanks, that a sample obtained in accordance with the applicable ASTM Standard has a total particulate contamination of less than 10 mg/l when tested in accordance with the applicable ASTM Standard.

d. At least once per ~~18~~ 24 months ~~during shutdown~~ by: L.6 LD.1

1. (Not Used).

SR 3.8.1.9 2. Verifying the diesel generator capability to reject a load of greater than or equal to its associated single largest post-accident load ~~1190 kW for diesel generator 0, greater than or equal to 638 kW for diesel generators 1A and 2A, and greater than or equal to 2421 kW for diesel generator 2B~~ while maintaining engine speed less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less. A.12 LA.5 A.6 A.11

Add proposed SR 3.8.1.9 NOTE

Add proposed SR 3.8.1.10 NOTE

SR 3.8.1.10 3. Verifying the diesel generator capability ~~to reject a load of~~ 2600 kW without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection. A.12

SR 3.8.1.11 4. Simulating or actual a loss of offsite power* by itself, and: L.12

SR 3.8.1.11 Note *All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, ~~as recommended by the manufacturer~~. LA.4

ITS 3.8.3

A.1

SURVEILLANCE REQUIREMENTS

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

SR3.8.3.3 7. Verifying the pressure in required diesel generator air start receivers to be greater than or equal to 200 psig.

Add proposed SR3.8.3.2

A.4

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day fuel tanks.

c. By sampling and analyzing stored and new fuel oil in accordance with the following:

1. At least once per 92 days, and for new fuel oil prior to addition to the storage tanks, that a sample obtained and tested in accordance with the applicable ASTM Standards has:

    a) A water and sediment content within applicable ASTM limits.

    b) A kinematic viscosity at 40°C within applicable ASTM limits.

2. At least every 31 days, and for new fuel oil prior to addition to the storage tanks, that a sample obtained in accordance with the applicable ASTM Standard has a total particulate contamination of less than 10 mg/l when tested in accordance with the applicable ASTM Standard.

moved to ITS Section 5.5

A.4

d. At least once per 18 months during shutdown by:

1. (Not Used).

2. Verifying the diesel generator capability* to reject a load of greater than or equal to 1190 kW for diesel generator 0, greater than or equal to 638 kW for diesel generators 1A and 2A, and greater than or equal to 2421 kW for diesel generator 2B while maintaining engine speed less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability* to reject a load of 2600 kW without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection.

4. Simulating a loss of offsite power* by itself, and:

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*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.

(See ITS 3.8.1)

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A.1

ITS 5.5

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

A.7

add proposed ITS 5.5.10

See ITS 3.8.1

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in required diesel generator air start receivers to be greater than or equal to 200 psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day fuel tanks.

5.5.10.a,b,c c. By sampling and analyzing stored and new fuel oil in accordance with the following: L.1

5.5.10.a 1. ~~At least once per 92 days, and~~ for new fuel oil prior to addition to the storage tanks, that a sample obtained and tested in accordance with the applicable ASTM Standards has: M.2

add proposed ITS 5.5.10.a.1

a) A water and sediment content within applicable ASTM limits. or a clear and bright appearance with proper color

b) A kinematic viscosity at 40°C and flash point M.2 within applicable ASTM limits.

add proposed ITS 5.5.10.b

2. At least every 31 days, ~~and for new fuel oil prior to addition to the storage tanks~~ that a sample obtained in accordance with the applicable ASTM Standard has a total particulate contamination of ~~less than~~ ≤ 10 mg/l when tested in accordance with the applicable ASTM Standard. L.1

d. At least once per 18 months during shutdown by:

1. (Not Used).

2. Verifying the diesel generator capability* to reject a load of greater than or equal to 1190 kW for diesel generator 0, greater than or equal to 638 kW for diesel generators 1A and 2A, and greater than or equal to 2421 kW for diesel generator 2B while maintaining engine speed less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability* to reject a load of 2600 kW without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection.

4. Simulating a loss of offsite power* by itself, and:

See ITS 3.8.1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies. A.7

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*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.

See ITS 3.8.1

ITS 3.8.1

A.1

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.11 a) For Divisions 1 and 2 and for Unit 1 Division 2:

1) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 13 seconds, energizes the auto-connected loads and operates for greater than or equal to 5 minutes while its generator is so loaded. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 150 volts and 60 ± 1.2 Hz during this test.

b) For Division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is so loaded. After energization, the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 150 volts and 60 ± 1.2 Hz during this test. M.1

or actual L.12

SR 3.8.1.12 5. Verifying that on an ECCS actuation test signal, without loss of offsite power, diesel generators 0, 2A, and 2B start* on the auto-start signal and operate on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ~~+ 416~~, -150 volts and 60 ~~+ 3.0~~, -1.2 Hz within 13 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within ~~these~~ limits during this test. 4160 ± 150 and 60 ± 1.2 L.9

SR 3.8.1.19 6. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal,* and: Actual or L.12

a) For Divisions 1 and 2:

1) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

SR 3.8.1.12 Note and SR 3.8.1.19 Note *All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, ~~as recommended by the manufacturer~~. LA.4

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ITS 3.8.1

A.1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.19

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 13 seconds, energizes the auto-connected emergency loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ±416 volts and 60 ±1.2 Hz during this test. LA.6

b) For Division 3:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ±416 volts and 60 ±1.2 Hz during this test. M.1

SR 3.8.1.13

7. Verifying that all diesel generator 0, 2A, and 2B automatic trips except the following are automatically bypassed on an actual or simulated ECCS actuation signal: L.12

a) For Divisions 1 and 2 - engine overspeed, generator differential current, ~~and emergency manual stop.~~ A.13 A.11

b) For Division 3 - engine overspeed, generator differential current, ~~and emergency manual stop.~~ A.12

SR 3.8.1.14

8. *Verifying the diesel generator operates** for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2860 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2400 kW to 2600 kW.*** The generator voltage and frequency shall be 4160 +420, -150 volts and 60 +3.0, -1.2 Hz within 13 seconds after the start signal; the steady-state L.13

Add proposed SR 3.8.1.14 Note 3

Add power factor requirement M.10

~~*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.~~ A.12

SR 3.8.1.14 Note 1
SR 3.8.1.15 Note 1
***Transients, outside of this load band, do not invalidate the surveillance tests.

ITS 3.8.1

A.1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

Add proposed SR 3.8.1.15 Note 3 A.11

SR 3.8.1.15 ~~generator voltage and frequency shall be maintained within these limits during this test.~~ L.13 Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.8.1.1.2.a.4.** A.12

~~9. Verifying* that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 2860 kW.~~ LA.7

SR 3.8.1.16
10. Verifying the diesel generator's capability~~*~~ to: A.12

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

SR 3.8.1.17
11. Verifying that with diesel generator 0, 2A, and 2B operating~~*~~ A.12 in a test mode and connected to its bus:

a) For Divisions 1 and 2, that a actual or simulated ECCS actuation signal overrides the test mode by returning the diesel generator to standby operation. L.12

b) For Division 3, that a actual or simulated trip of the diesel generator overcurrent relay trips the SAT feed breaker to bus 243 and that the diesel generator continues to supply normal bus loads.

SR 3.8.1.18
12. Verifying that the ~~automatic load sequence timer~~ time delay relay is OPERABLE with the interval between each load block within ±10% of its design interval for diesel generators 0 and 2A. A.14

~~13. Verifying that the following diesel generator lockout features prevent diesel generator operation only when required:~~ L.14

SR3.8.1.15 Note 2 *All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, ~~as recommended by the manufacturer.~~ LA.4

SR 3.8.1.15 Note 1 **If Surveillance Requirement 4.8.1.1.2.a.4 is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at ≥ 2400 kw and ≤ 2600 kW for 2 hours ~~or until operating temperature has stabilized.~~ Momentary transients below the load limit do not invalidate the test. A.15

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A.1

ITS 3.8.1

a) ~~Generator underfrequency.~~
b) ~~Low lube oil pressure.~~
c) ~~High jacket cooling temperature.~~
d) ~~Generator reverse power.~~
e) ~~Generator overcurrent.~~
f) ~~Generator loss of field.~~
g) ~~Engine cranking lockout.~~

L.14

L.15

L.16

SR 3.8.1.20

e. At least once per 10 years ~~or after any modifications which could affect diesel generator interdependence~~ by starting diesel generators 0, 2A, and 2B simultaneously*, ~~during shutdown~~, and verifying that all three diesel generators accelerate to ~~900 rpm + 5, -2%~~ ≥58.8 Hz in less than or equal to 13 seconds. Add proposed voltage limit

L.9

M.11

f. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND, of the ASME Code in accordance with ASME Code Section 11, Article IWD-5000.

A.3

Moved to ITS 3.8.3

4.8.1.1.3 Reports - (Not Used).

Add proposed SR 3.8.1.21

A.18

SR3.8.1.20 Note

*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, ~~as recommended by the manufacturer~~.

LA.4

A.1

ITS 3.8.3

a) Generator underfrequency.
b) Low lube oil pressure.
c) High jacket cooling temperature.
d) Generator reverse power.
e) Generator overcurrent.
f) Generator loss of field.
g) Engine cranking lockout.

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting diesel generators 0, 2A, and 2B simultaneously*, during shutdown, and verifying that all three diesel generators accelerate to 900 rpm + 5, -2% in less than or equal to 13 seconds.

~~f. At least once per 10 years by:~~

~~1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and~~

~~2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND, of the ASME Code in accordance with ASME Code Section 11, Article IWD-5000.~~

L3

4.8.1.1.3 Reports - (Not Used)

(See ITS 3.8.1)

*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.

A.1

ELECTRICAL POWER SYSTEMS

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

M.1

LCO 3.8.2 3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE: M.2

LCO 3.8.2.a a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

LCO 3.8.2.b LCO 3.8.2.c LCO 3.8.2.d b. Diesel generator 0 or 2A, and diesel generator 2B when the HPCS system is required to be OPERABLE, and diesel generator 1A when the offsite power source for standby gas treatment system subsystem A or control room and auxiliary electric equipment room emergency filtration system train A is inoperable and either or both systems are required to be OPERABLE, with each diesel generator having:

SR 3.8.2.1 1. For diesel generator 0, 1A, and 2A:

a) A separate day fuel tank containing a minimum of 250 gallons of fuel. A.2

b) A separate fuel storage system containing a minimum of 31,000 gallons of fuel. A.3 moved to ITS 3.8.3

SR 3.8.2.1 2. For diesel generator 2B, a separate fuel storage tank/day tank ~~containing a minimum of~~ ~~29,750 gallons of fuel~~. A.2 550 gallons of fuel M.3

A.2

~~3. A fuel transfer pump.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *. A.4 M.1

ACTION: Add proposed ACTION A Note

Add Proposed Required Action A.1

ACTIONS A and B a. With all offsite circuits inoperable and/or with diesel generators 0 or 2A inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. M.4 Add Required Actions A.2.4 and B.4

ACTION C b. With diesel generator 2B inoperable, restore the inoperable diesel generator 2B to OPERABLE status within 72 hours or declare the HPCS system inoperable ~~and take the ACTION required by Specifications 3.5.2 and 3.5.3.~~ A.5

ACTION D c. With diesel generator 1A inoperable, declare standby gas treatment system subsystem A and control room and auxiliary electric equipment room emergency filtration system train A inoperable ~~and take the ACTION required by Specifications 3.6.5.3 and 3.7.2.~~ A.5

ACTIONS NOTE d. The provisions of Specification 3.0.3 are not applicable.

Applicability *When handling irradiated fuel in the secondary containment.

add proposed fuel oil storage tank and starting air LCO A.2

ELECTRICAL POWER SYSTEMS

A.C. SOURCES - SHUTDOWN

A.1

ITS 3.8.3

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator 0 or 2A, and diesel generator 2B when the HPCS system is required to be OPERABLE, and diesel generator 1A when the offsite power source for standby gas treatment system subsystem A or control room and auxiliary electric equipment room emergency filtration system train A is inoperable and either or both systems are required to be OPERABLE, with each diesel generator having:

1. For diesel generator 0, 1A, and 2A:

   a) A separate day fuel tank containing a minimum of 250 gallons of fuel.

   b) A separate fuel storage system containing a minimum of 31,000 gallons of fuel.

SR 3.8.3.1

2. For diesel generator 2B, a separate fuel storage tank/day tank containing a minimum of 29,750 gallons of fuel. A.3

3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *. A.2

ACTION:

a. With all offsite circuits inoperable and/or with diesel generators 0 or 2A inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With diesel generator 2B inoperable, restore the inoperable diesel generator 2B to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

c. With diesel generator 1A inoperable, declare standby gas treatment system subsystem A and control room and auxiliary electric equipment room emergency filtration system train A inoperable and take the ACTION required by Specifications 3.6.5.3 and 3.7.2.

d. The provisions of Specification 3.0.3 are not applicable.

*When handling irradiated fuel in the secondary containment. A.2

add proposed ACTIONS A, B, C, D, and E and ACTIONS NOTE L.1

See ITS 3.8.2

ITS 3.8.2

A.1

ELECTRICAL POWER SYSTEMS

add proposed SR 3.8.2.1 Note 1 — L.1

SURVEILLANCE REQUIREMENTS

add proposed SR 3.8.2.1 Note 2 — L.2

SR 3.8.2.1

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

A.6

SR 3.8.2.1 Note 1

Movement of certain requirements to ITS 3.8.3 — A.3

Add exception to CTS SR 4.8.1.1.1.b (i.e., ITS SR 3.8.1.8)

Add exception to CTS SR 4.8.1.1.2.d.11 (i.e., ITS SR 3.8.1.17)

Add exception to CTS SR 4.8.1.1.2.e (i.e., ITS SR 3.8.1.20)

A.7

A.1

ITS 3.8.3

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1
SR 3.8.3.2
SR 3.8.3.3

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1; 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

portions not applicable to fuel oil or starting air

(See ITS 3.8.2)

A.1

ITS 3.8.1

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A. C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. distribution system electrical divisions shall be OPERABLE and energized:

a. Division 1, consisting of;

1. 4160-volt bus 241Y.
2. 480-volt busses 235X and 235Y.
3. 480-volt MCCs 235X-1, 235X-2, 235X-3, 235Y-1 and 235Y-2.
4. 120-volt A.C. distribution panels in 480-volt MCCs 235X-1, 235X-2, 235X-3 and 235Y-1.

See ITS 3.8.7

b. Division 2, consisting of;

1. 4160-volt bus 242Y.
2. 480-volt busses 236X and 236Y.
3. 480-volt MCCs 236X-1, 236X-2, 236X-3, 236Y-1 and 236Y-2.
4. 120-volt A.C. distribution panels in 480-volt MCCs 236X-1, 236X-2, 236X-3 and 236Y-2.

c. Division 3, consisting of;

1. 4160-volt bus 243.
2. 480-volt MCC 243-1.
3. 120-volt A.C. distribution panels in 480-volt MCC 243-1.

LA.7

LCO 3.8.1.a

d. Unit 1 Division 1, consisting of;

1. 4160-volt bus 141Y.
2. Breaker 1414 OPERABLE ~~or~~ closed.

LA.1

e. Unit 1 Division 2, consisting of;

1. 4160-volt bus 142Y.
2. 480-volt busses 136X and 136Y.
3. 480-volt MCCs 136X-1, 136X-2, 136X-3, 136Y-1, and 136Y-2.
4. 120-volt A.C. distribution panels in 480 volt MCCs 136X-1, 136X-2, 136X-3, and 136Y-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

See ITS 3.8.7

ITS 3.8.7

A.1

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A. C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.8.7

3.8.2.1 The following A.C. distribution system electrical divisions shall be OPERABLE and energized: LA.1

a. Division 1, ~~consisting of;~~

~~1. 4160-volt bus 241Y.~~
~~2. 480-volt busses 235X and 235Y.~~
~~3. 480-volt MCCs 235X-1, 235X-2, 235X-3, 235Y-1 and 235Y-2.~~
~~4. 120-volt A.C. distribution panels in 480-volt MCCs 235X-1, 235X-2, 235X-3 and 235Y-1.~~

b. Division 2, ~~consisting of;~~

~~1. 4160-volt bus 242Y.~~
~~2. 480-volt busses 236X and 236Y.~~
~~3. 480-volt MCCs 236X-1, 236X-2, 236X-3, 236Y-1 and 236Y-2.~~
~~4. 120-volt A.C. distribution panels in 480-volt MCCs 236X-1, 236X-2, 236X-3 and 236Y-2.~~

LA.1

c. Division 3, ~~consisting of;~~

~~1. 4160-volt bus 243.~~
~~2. 480-volt MCC 243-1.~~
~~3. 120-volt A.C. distribution panels in 480-volt MCC 243-1.~~

d. Unit 1 Division 1, consisting of;

1. 4160-volt bus 141Y.
2. Breaker 1414 OPERABLE or closed.

A.2
moved to ITS 3.8.1

e. Unit 1 Division 2, ~~consisting of;~~

~~1. 4160-volt bus 142Y.~~
~~2. 480-volt busses 136X and 136Y.~~
~~3. 480-volt MCCs 136X-1, 136X-2, 136X-3, 136Y-1, and 136Y-2.~~
~~4. 120-volt A.C. distribution panels in 480 volt MCCs 136X-1, 136X-2, 136X-3, and 136Y-2.~~

LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

add proposed description of equipment required to be supported by opposite unit bus A.5

A.1

ITS 3.8.1

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

See ITS 3.8.7

a. With either Division 1 or Division 2 of the above required A.C. distribution system inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours

b. With Division 3 of the above required A.C. distribution system inoperable or not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

A.17

c. ACTION A — With Unit 1 Division 1 or Unit 1 Division 2 of the above required A.C. distribution systems inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 7 days ACTION G — or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

See ITS 3.8.7

d. With both Unit 1 Division 1 and Unit 1 Division 2 of the above required A.C. distribution systems inoperable or not energized, restore at least one of the inoperable A.C. distribution systems to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The above required A.C. distribution system electrical divisions shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/panels.

ITS 3.8.7

A.1

L.1

ACTION:

LA.1

add proposed Second Completion time

M.1

ACTION A a. With either Division 1 or Division 2 of the above required A.C. distribution system inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 8 hours

ACTION D or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours

LA.1 M.4 A.3

ACTION E b. With Division 3 of the above required A.C. distribution system inoperable or not energized, declare the HPCS system inoperable ~~and take the ACTION required by Specification 3.5.1.~~

A.2

ACTION C c. With Unit 1 Division 1 or Unit 1 Division 2 of the above required A.C. distribution systems inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 7 days

moved to ITS 3.8.1

ACTION D or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

LA.1

~~d. With both Unit 1 Division 1 and Unit 1 Division 2 of the above required A.C. distribution systems inoperable or not energized, restore at least one of the inoperable A.C. distribution systems to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

L.2

add proposed Action G

M.2

L.1

SURVEILLANCE REQUIREMENTS

LA.1

SR 3.8.7.1 4.8.2.1 The above required A.C. distribution system electrical divisions shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/panels.

A.1

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO3.8.8 3.8.2.2 As a minimum, Division 1 or Division 2, and Division 3 when the HPCS system is required to be OPERABLE, and Unit 1 Division 2 when the standby gas treatment system and/or the control room and auxiliary electric equipment room emergency filtration system are required to be OPERABLE, of the A.C. distribution system shall be OPERABLE ~~and energized~~ with:

to support equipment required to be OPERABLE

M.1

LA.1

~~a. Division 1, consisting of;~~

~~1. 4160-volt bus 241Y.~~
~~2. 480-volt busses 235X and 235Y.~~
~~3. 480-volt MCCs 235X-1, 235X-2, 235X-3, 235Y-1, and 235Y-2.~~
~~4. 120-volt A.C. distribution panels in 480-volt MCCs 235X-1, 235X-2, 235X-3, and 235Y-1.~~

~~b. Division 2, consisting of;~~

~~1. 4160-volt bus 242Y.~~
~~2. 480-volt busses 236X and 236Y.~~
~~3. 480-volt MCCs 236X-1, 236X-2, 236X-3, 236Y-1, and 236Y-2.~~
~~4. 120-volt A.C. distribution panels in 480-volt MCCs 236X-1, 236X-2, 236X-3, and 236Y-2.~~

LA.1

~~c. Division 3, consisting of;~~

~~1. 4160-volt bus 243~~
~~2. 480-volt MCC 243-1.~~
~~3. 120-volt A.C. distribution panels in 480-volt MCC 243-1.~~

~~d. Unit 1 Division 2, consisting of;~~

~~1. 4160-volt bus 142Y.~~
~~2. 480-volt busses 136X and 136Y.~~
~~3. 480-volt MCCs 136X-1, 136X-2, 136X-3, and 136Y-1.~~
~~4. 120-volt A.C. distribution panels in 480-volt MCCs 136X-1, 136X-2, and 136X-3.~~

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

APPLICABILITY *When handling irradiated fuel in the secondary containment.

A.1

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

M.1

ACTION: one or more required   add proposed Required Action A.1

ACTION A

a. With both Division I and 2 of the above required A.C. distribution system inoperable ~~or not energized~~, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. ← add proposed Required Actions A.2.4 and A.2.5   M.2

b. With Division 3 of the above required A.C. distribution system inoperable ~~or not energized~~, declare the HPCS system inoperable ~~and take the ACTION required by Specifications 3.5.2 and 3.5.3.~~ A.2   add proposed Required Actions A.2.1, A.2.2, and A.2.3   M.3

c. With Unit I Division 2 of the above required A.C. distribution system inoperable ~~or not energized~~, declare the standby gas treatment system subsystem A and control room and auxiliary electric equipment room emergency filtration system train A inoperable ~~and take the ACTION required by Specifications 3.6.5.3 and 3.7.2.~~ A.2   add proposed Required Action A.2.4   M.2

Note to ACTIONS

d. The provisions of Specification 3.0.3 are not applicable.

add proposed Required Actions A.2.1, A.2.2., A.2.3, and A.2.4   M.2

LA.1

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1   4.8.2.2 At least the above required A.C. distribution system electrical division(s) shall be determined OPERABLE ~~and/energized~~ at least once per 7 days by verifying correct breaker alignment and voltage on the busses/panels. LA.2

ITS 3.8.4

A.1

<General Description> A.2

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.8.4 A.2

3.8.2.3 The following D.C. distribution system electrical divisions shall be OPERABLE and energized:

LA.1

a. Division 1~~, consisting of:~~
   1. ~~125-volt battery 2A.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 211Y.

b. Division 2~~, consisting of:~~
   1. ~~125-volt battery 2B.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 212Y.

c. Division 3~~, consisting of:~~
   1. ~~125-volt battery 2C.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 213.

d. ~~Unit 1~~ Division 2~~, consisting of:~~
   1. ~~125-volt battery 1B.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 112Y.

See ITS 3.8.7

A.1 opposite unit

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

ACTION A
ACTION E

a. With either Division 1 or Division 2 inoperable ~~or not energized~~, restore the inoperable division to OPERABLE ~~and energized status~~ within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

See ITS 3.8.7

ACTION B

b. With Division 3 inoperable ~~or not energized~~, declare the HPCS system inoperable ~~and take the ACTION required by Specification 3.5.1~~.

A.3

ACTION D
ACTION E

c. With ~~Unit 1~~ opposite Unit Division 2 inoperable ~~or not energized~~, restore the inoperable division to OPERABLE ~~and energized status~~ within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

A.1

See ITS 3.8.7

Page 6 of 10

A.1

ITS 3.8.6

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

<General Arrangement> A.2

add proposed LCO 3.8.6

3.8.2.3 The following D.C. distribution system electrical divisions shall be OPERABLE and energized:

a. Division 1, consisting of;
   1. 125-volt battery 2A.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 211Y.

b. Division 2, consisting of;
   1. 125-volt battery 2B.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 212Y.

c. Division 3, consisting of;
   1. 125-volt battery 2C.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 213.

d. Unit 1 Division 2, consisting of;
   1. 125-volt battery 1B.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 112Y.

<See ITS 3.8.4 and ITS 3.8.7>

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. A.3

ACTION:

a. With either Division 1 or Division 2 inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division 3 inoperable or not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

c. With Unit 1 Division 2 inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

<See ITS 3.8.4 and ITS 3.8.7>

ITS 3.8.7

A.1

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. distribution system electrical divisions shall be OPERABLE and energized: LA.1

a. Division 1, consisting of: LA.1
   1. 125-volt battery 2A.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 211Y.

b. Division 2, consisting of: LA.1
   1. 125-volt battery 2B.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 212Y.

c. Division 3, consisting of: LA.1
   1. 125-volt battery 2C.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 213.

d. Unit 1 Division 2, consisting of: LA.1
   1. 125-volt battery 1B.
   2. 125-volt full capacity charger.
   3. 125-volt distribution panel 112Y. LA.1

See ITS 3.8.4

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

L.1

M.1 add proposed second Completion time

Action B / Action D — a. With either Division 1 or Division 2 inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. LA.1

Action E — b. With Division 3 inoperable or not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1. A.3

Action C / Action D — c. With Unit 1 Division 2 inoperable or not energized, restore the inoperable division to OPERABLE and energized status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. LA.1

add proposed description of equipment required to be supported by opposite unit bus A5

add proposed ACTION G M2 L.1

A.1

ITS 3.8.4

SURVEILLANCE REQUIREMENTS

⟨See ITS 3.8.7⟩

4.8.2.3.1 Each of the above required D.C. distribution system electrical divisions shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 125 volts.

4.8.2.3.2 Each 125-volt battery and charger shall be demonstrated OPERABLE:

SR3.8.4.1

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8.2.3.2-1 meet the Category A limits, and — A.2 — moved to ITS 3.8.6

   2. Total battery terminal voltage is greater than or equal to 128 volts on float charge.

SR3.8.4.2

b. At least once per 92 days and within 7 days after a battery discharge with battery voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that: — L.1

   1. The parameters in Table 4.8.2.3.2-1 meet the Category B limits, — moved to ITS 3.8.6 — A.2

   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohm, and

   3. The average electrolyte temperature of at least 10 connected cells is above 60°F. — A.2 — moved to ITS 3.8.6

c. At least once per 18 (24) months by verifying that: — LD.1

SR3.8.4.3

   1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

SR 3.8.4.4

   2. The cell-to-cell and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material, — L.2

SR3.8.4.5

   The resistance of each cell and terminal connection is less than or equal to 150 x 10^{-6} ohm, and

SR3.8.4.6

   4. The battery charger will supply a load equal to the manufacturer's rating for at least 8 (4) hours. — A.4 — L.7

ITS 3.8.6

A.1

ELECTRICAL POWER SYSTEMS

<See ITS 3.8.4>

SURVEILLANCE REQUIREMENTS

<See ITS 3.8.7>

4.8.2.3.1 Each of the above required D.C. distribution system electrical divisions shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 125 volts.

4.8.2.3.2 Each 125-volt battery and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

SR 3.8.6.1 1. The parameters in Table 4.8.2.3.2-1 meet the Category A limits, and

2. Total battery terminal voltage is greater than or equal to 128 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that: L.1

SR 3.8.6.2 1. The parameters in Table 4.8.2.3.2-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohm, and

SR 3.8.6.3 3. The average electrolyte temperature of ~~at least 10~~ connected cells is above 60°F.

≥

representative LA.1

L.7

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material,

3. The resistance of each cell and terminal connection is less than or equal to 150 x 10^{-6} ohm, and

4. The battery charger will supply a load equal to the manufacturer's rating for at least 8 hours.

<See ITS 3.8.4>

ITS 3.8.7

A.1

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1 4.8.2.3.1 Each of the above required D.C. distribution system electrical divisions shall be determined OPERABLE ~~and energized~~ at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 125 volts. LA.1 LA.2

4.8.2.3.2 Each 125-volt battery and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8.2.3.2-1 meet the Category A limits, and

2. Total battery terminal voltage is greater than or equal to 128 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

1. The parameters in Table 4.8.2.3.2-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohm, and

3. The average electrolyte temperature of at least 10 connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material,

3. The resistance of each cell and terminal connection is less than or equal to 150 x 10^{-6} ohm, and

4. The battery charger will supply a load equal to the manufacturer's rating for at least 8 hours.

<See ITS 3.8.4>

SURVEILLANCE REQUIREMENTS (Continued)

24 LD.1

SR3.8.4.7 d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the 240 minute design cycle when the battery is subjected to a battery service test. L.3 LA.2

for the design duty cycle

L.3

SR3.8.4.8 e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturers rating when subjected to a performance discharge test. Once per 60-month interval,

Note to SR 3.8.4.7 This performance discharge test may be performed in lieu of the battery service test. provided it envelops the service test L.4

modified L.4

or modified performance discharge test L.5

SR3.8.4.8 f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

LA3

with capacity < 100% of manufacturer's rating

add proposed 3rd frequency L.6

add proposed SR 3.8.4.9 and Notes to SRs

A.5

A.1

ITS 3.8.4

A.2 moved to ITS 3.8.6

TABLE 4.8.2.3.2-1

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A(1)	CATEGORY B(2)	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable (3) value for each connected cell
Electrolyte Level	>Minimum level indication mark and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	≥ 1.200(b)	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amperes when on float charge.
(c) May be corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 7 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

ITS 3.8.6

A.1

Table 3.8.6-1

TABLE 4.8.2.3.2-1

BATTERY SURVEILLANCE REQUIREMENTS

	Category A	Category B	Category C
	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable (3) value for each connected cell
Electrolyte Level	>Minimum level indication mark and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts ~~(c)~~ M.1	> 2.07 volts
Specific Gravity(a)	$\geq$ 1.200 ⓑ	$\geq$ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195 ⓑ

add proposed footnote (a) L.2

L.3

Note (b) (a) Corrected for electrolyte temperature and level.

Note (c) (b) Or battery charging current is less than 2 amperes when on float charge. add proposed footnote (c) time allowance M.2

~~(c) May be corrected for average electrolyte temperature.~~ M.1

ACTION A (1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next ⑦ days. ㉛ L4

add proposed Required Action A.1 M.3

ACTION A (2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within ⑦ days. ㉛ L4

ACTION B (3) Any Category B parameter not within its allowable value indicates an inoperable battery.

add proposed Action B for electrolyte temperature and category A or B limits not restored A.5

add proposed ACTIONS note A.4

ITS 3.8.5

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - SHUTDOWN A.1

LIMITING CONDITION FOR OPERATION <General Description> A.2

~~3.8.2.4 As a minimum, Division 1 or Division 2, and Division 3 when the HPCS system is required to be OPERABLE, and Unit 1 Division 2 when the standby gas treatment system and/or the control room and auxiliary electric equipment room emergency filtration system are required to be OPERABLE, of the D.C. distribution system shall be OPERABLE and energized with:~~

DC electrical power subsystem(s) shall be OPERABLE to support the electrical power subsystems required by LCO 3.8.8, "Distribution Systems - Shutdown" A.2 M.1

a. ~~Division 1, consisting of;~~
   1. ~~125-volt battery 2A.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 211Y.

b. ~~Division 2, consisting of;~~
   1. ~~125-volt battery 2B.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 212Y.

c. ~~Division 3, consisting of;~~
   1. ~~125-volt battery 2C.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 213.

d. ~~Unit 1 Division 2, consisting of;~~
   1. ~~125-volt battery 1B.~~
   2. ~~125-volt full capacity charger.~~
   3. 125-volt distribution panel 112Y.

LA.1

See ITS 3.8.8

LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION: one or more required M.1

add proposed Required Action B.1

ACTION B a. With both Division 1 distribution panel 211Y and Division 2 distribution panel 212Y of the above required D.C. distribution system inoperable or not energized, suspend CORE ALTERATIONS, handling of irradiated fuel cask in the secondary containment and operations with a potential for draining the reactor vessel.

See ITS 3.8.8

ACTION B b. With Division 3 distribution panel 213 of the above required D.C. distribution system inoperable or not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3. A.3

add proposed Required Actions B.2.1 and B.2.2 M3

add proposed Required Action B.2.4 M.2

---

*When handling irradiated fuel in the secondary containment.

APPLICABILITY

A.1

ITS 3.8.6

ELECTRICAL POWER SYSTEMS

<General Arrangement> A.2

D.C. DISTRIBUTION - SHUTDOWN

add proposed LCO 3.8.6

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, Division 1 or Division 2, and Division 3 when the HPCS system is required to be OPERABLE, and Unit 1 Division 2 when the standby gas treatment system and/or the control room and auxiliary electric equipment room emergency filtration system are required to be OPERABLE, of the D.C. distribution system shall be OPERABLE and energized with:

a. Division 1, consisting of;

1. 125-volt battery 2A.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 211Y.

b. Division 2, consisting of;

1. 125-volt battery 2B.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 212Y.

c. Division 3, consisting of;

1. 125-volt battery 2C.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 213.

d. Unit 1 Division 2, consisting of;

1. 125-volt battery 1B.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 112Y.

<See ITS 3.8.5 and 3.8.8>

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *. A.3

ACTION:

a. With both Division 1 distribution panel 211Y and Division 2 distribution panel 212Y of the above required D.C. distribution system inoperable or not energized, suspend CORE ALTERATIONS, handling of irradiated fuel cask in the secondary containment and operations with a potential for draining the reactor vessel.

b. With Division 3 distribution panel 213 of the above required D.C. distribution system inoperable or not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

---

*When handling irradiated fuel in the secondary containment. A.3

ITS 3.8.8

A.1

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.8.8

3.8.2.4 As a minimum, Division 1 or Division 2, and Division 3 when the HPCS system is required to be OPERABLE, and Unit 1 Division 2 when the standby gas treatment system and/or the control room and auxiliary electric equipment room emergency filtration system are required to be OPERABLE, of the D.C. distribution system shall be OPERABLE ~~and energized~~ with:

to support equipment required to be OPERABLE

M.1

LA.1

a. Division 1, consisting of;

1. 125-volt battery 2A.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 211Y.

LA.1

b. Division 2, consisting of;

1. 125-volt battery 2B.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 212Y.

LA.1

c. Division 3, consisting of;

1. 125-volt battery 2C.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 213.

LA.1

d. Unit 1 Division 2, consisting of;

1. 125-volt battery 1B.
2. 125-volt full capacity charger.
3. 125-volt distribution panel 112Y.

LA.1

(See ITS 3.8.5)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION: One or more required

M.1

add proposed Required Action A.1

Action A

a. With both Division 1 distribution panel 211Y and Division 2 distribution panel 212Y of the above required D.C. distribution system inoperable ~~or not energized~~, suspend CORE ALTERATIONS, handling of irradiated fuel cask in the secondary containment and operations with a potential for draining the reactor vessel. ← add proposed Required Actions A.2.4 and A.2.5

LA.1

M.2

b. With Division 3 distribution panel 213 of the above required D.C. distribution system inoperable ~~or not energized~~, declare the HPCS system inoperable/ ~~and take the ACTION required by Specifications 3.5.2 and 3.5.3~~.

LA.1

A.2

add proposed Required Actions A.2.1, A.2.2, and A.2.3 M3

add proposed Required Action A.2.4 M.2

Applicability *When handling irradiated fuel in the secondary containment.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued) Verify within 1 hour L2 Add proposed Condition A Note M.4

Action A c. With one division battery and/or battery charger inoperable, operation may continue provided the Unit tie breakers for the affected division are OPERABLE and aligned to supply power to the affected distribution panel from the associated OPERABLE Unit 1 125-volt D.C. distribution panel; restore the inoperable battery and/or charger to OPERABLE status within 72 hours or declare the division

Action B distribution panel inoperable.

A.4

Add proposed Required Actions B.2.1, B.2.2, and B.2.3

See ITS 3.8.8

Action B d. With Unit 1 Division 2 of the above required D.C. distribution system inoperable or not energized, declare the standby gas treatment system subsystem A and the control room and auxiliary electric equipment room emergency filtration system train A inoperable and take the ACTION required by Specifications 3.6.5.3 and 3/7.2. A3

Note to Action B e. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 At least the above required D.C. distribution system electrical division(s) shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the panel(s) with an overall voltage of greater than or equal to 125 volts. See ITS 3.8.8

add proposed Note L.1

SR 3.8.5.1 4.8.2.4.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

ITS 3.8.6

A.1

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one division battery and/or battery charger inoperable, operation may continue provided the Unit tie breakers for the affected division are OPERABLE and aligned to supply power to the affected distribution panel from the associated OPERABLE Unit 1 125-volt D.C. distribution panel; restore the inoperable battery and/or charger to OPERABLE status within 72 hours or declare the division distribution panel inoperable.

d. With Unit 1 Division 2 of the above required D.C. distribution system inoperable or not energized, declare the standby gas treatment system subsystem A and the control room and auxiliary electric equipment room emergency filtration system train A inoperable and take the ACTION required by Specifications 3.6.5.3 and 3.7.2.

e. The provisions of Specification 3.0.3 are not applicable.

<See ITS 3.8.5 and ITS 3.8.8>

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 At least the above required D.C. distribution system electrical division(s) shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the panel(s) with an overall voltage of greater than or equal to 125 volts.

~~4.8.2.4.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.~~ A.4

<See ITS 3.8.8>

ITS 3.8.8

A.1

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one division battery and/or battery charger inoperable, operation may continue provided the Unit tie breakers for the affected division are OPERABLE and aligned to supply power to the affected distribution panel from the associated OPERABLE Unit 1 125-volt D.C. distribution panel; restore the inoperable battery and/or charger to OPERABLE status within 72 hours or declare the division distribution panel inoperable.

See ITS 3.8.5

A.3

LA.1

ACTION A

d. With Unit 1 Division 2 of the above required D.C. distribution system inoperable or not energized, declare the standby gas treatment system subsystem A and the control room and auxiliary electric equipment room emergency filtration system train A inoperable and take the ACTION required by Specifications 3.6.5.3 and 3.7.2.

A.2

add proposed Required Actions A.2.1, A.2.2, A.2.3 and A.2.4

Note to ACTIONS

e. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

4.8.2.4.1 At least the above required D.C. distribution system electrical division(s) shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the panel(s) with an overall voltage of greater than or equal to 125 volts.

LA.2

4.8.2.4.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

See ITS 3.8.5

Page 8 of 8

CTS 3/4.8.3.1

ELECTRICAL POWER SYSTEMS

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.3.1 At least the following A.C. circuits inside primary containment shall be deenergized*:

a. Installed welding grid systems 2A and 2B, and

b. All drywell lighting circuits.

c. All drywell hoists and cranes circuits.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

R.1

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required A.C. circuits shall be determined to be de-energized at least once per 24 hours** by verifying that the associated circuit breakers are in the tripped condition.

---

*Except during entry into the drywell.

**Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

Page 2 of 2

CTS 3/4.8.3.2

R.1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

~~LIMITING CONDITION FOR OPERATION~~

~~3.8.3.2 Primary and backup primary containment penetration conductor overcurrent protective devices associated with each primary containment medium and high voltage (6.9 kV, 4.16 kV and 480 volt) electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating.~~

~~APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.~~

~~ACTION:~~

~~With one or more of the primary containment penetration conductor overcurrent protective devices inoperable, restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.8.3.2 Each of the primary containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:~~

~~a. At least once per 18 months:~~

~~1. By verifying that the 6.9 kV and 4.16 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing:~~

~~a) A CHANNEL CALIBRATION of the associated protective relays, and~~

~~b) An integrated system functional test of the breakers overcurrent protective trip circuit which includes simulated automatic actuation of the trip system to demonstrate that the overall penetration protection design remains within operable limits.~~

~~c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.~~

Page 3 of 4

CTS 3/4.8.3.2

SURVEILLANCE REQUIREMENTS (Continued)

~~a. By selecting and functionally testing a representative sample of at least 10% of each type of 480-volt circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of 120% of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to 120% of a value specified for test current by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.~~

~~b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.~~

R.1

Page 4 of 4

CTS 3/4.8.3.3

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.3.3 The thermal overload protection of each valve shown in Table 3.8.3.3-1 shall be bypassed continuously or under accident conditions, as applicable, by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device, take administrative action to continuously bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.3.3.1 The thermal overload protection for the above required valves shall be verified to be bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions and by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing:

a. At least once per 18 months, and

b. Following maintenance on the motor starter.

LA.1

4.8.3.3.2 The thermal overload protection for the above required valves which are continuously bypassed shall be verified to be bypassed following testing during which the thermal overload protection was temporarily placed in force.

CTS 3/4.8.3.3

TABLE 3.8.3.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LA.1

	VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED
a.	1VG001	Accident Conditions	SBGTS
	1VG003	Accident Conditions	
	2VG001	Accident Conditions	
	2VG003	Accident Conditions	
b.	2VP113A	Accident Conditions	Primary containment chilled water coolers
	2VP113B	Accident Conditions	
	2VP114A	Accident Conditions	
	2VP114B	Accident Conditions	
	2VP053A	Accident Conditions	
	2VP053B	Accident Conditions	
	2VP063A	Accident Conditions	
	2VP063B	Accident Conditions	
c.	2VQ038	Accident Conditions	Primary containment vent and purge system
	2VQ032	Accident Conditions	
	2VQ035	Accident Conditions	
	2VQ047	Accident Conditions	
	2VQ048	Accident Conditions	
	2VQ050	Accident Conditions	
	2VQ051	Accident Conditions	
	2VQ068	Accident Conditions	
	2VQ037	Accident Conditions	
d.	2WR179	Accident Conditions	RBCCW system
	2WR180	Accident Conditions	
	2WR040	Accident Conditions	
	2WR029	Accident Conditions	
e.	2B21 - F067A	Accident Conditions	Main steam system
	2B21 - F067B	Accident Conditions	
	2B21 - F067C	Accident Conditions	
	2B21 - F067D	Accident Conditions	
	2B21 - F019	Accident Conditions	
	2B21 - F016	Accident Conditions	
	2B21 - F020	Continuous	
	2B21 - F068	Continuous	
	2B21 - F070	Continuous	
	2B21 - F069	Continuous	
	2B21 - F071	Continuous	
	2B21 - F072	Continuous	
	2B21 - F073	Continuous	
f.	2B21 - F065A	Continuous	Main feedwater system
	2B21 - F065B	Continuous	

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CTS 3/4.8.3.3

TABLE 3.8.3.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED
g.	2E21 - F001	Continuous	LPCS system
	2E21 - F005	Accident Conditions	
	2E21 - F011	Accident Conditions	
	2E21 - F012	Accident Conditions	
h.	2C41 - F001A	Accident Conditions	SBLCS
	2C41 - F001B	Accident Conditions	
i.	2G33 - F001	Accident Conditions	RWCU
	2G33 - F004	Accident Conditions	
j.	2E12 - F052A	Accident Conditions	RHR system
	2E12 - F064A	Accident Conditions	
	2E12 - F087A	Accident Conditions	
	2E12 - F004A	Continuous	
	2E12 - F047A	Continuous	
	2E12 - F048A	Accident Conditions	
	2E12 - F003A	Continuous	
	2E12 - F026A	Accident Conditions	
	2E12 - F068A	Continuous	
	2E12 - F073A	Continuous	
	2E12 - F074A	Continuous	
	2E12 - F011A	Accident Conditions	
	2E12 - F024A	Accident Conditions	
	2E12 - F016A	Accident Conditions	
	2E12 - F017A	Accident Conditions	
	2E12 - F027A	Accident Conditions	
	2E12 - F004B	Continuous	
	2E12 - F047B	Continuous	
	2E12 - F048B	Accident Conditions	
	2E12 - F003B	Continuous	
	2E12 - F068B	Continuous	
	2E12 - F073B	Continuous	
	2E12 - F074B	Continuous	
	2E12 - F026B	Accident Conditions	
	2E12 - F011B	Accident Conditions	
j.	2E12 - F024B	Accident Conditions	RHR system
	2E12 - F006B	Continuous	
	2E12 - F016B	Accident Conditions	
	2E12 - F017B	Accident Conditions	
	2E12 - F042B	Accident Conditions	
	2E12 - F064B	Accident Conditions	
	2E12 - F093	Continuous	
	2E12 - F021	Accident Conditions	
	2E12 - F004C	Continuous	
	2E12 - F052B	Accident Conditions	
	2E12 - F087B	Accident Conditions	

LA.1

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CTS 3/4.8.3.3

TABLE 3.8.3.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LA.1

	VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED
	2E12 - F099B	Accident Conditions	
	2E12 - F099A	Accident Conditions	
	2E12 - F008	Accident Conditions	
	2E12 - F009	Accident Conditions	
	2E12 - F040A	Accident Conditions	
	2E12 - F040B	Accident Conditions	
	2E12 - F049A	Accident Conditions	
	2E12 - F049B	Accident Conditions	
	2E12 - F053A	Accident Conditions	
	2E12 - F053B	Accident Conditions	
	2E12 - F006A	Continuous	
	2E12 - F023	Accident Conditions	
	2E12 - F027B	Accident Conditions	
	2E12 - F042A	Accident Conditions	
	2E12 - F042C	Accident Conditions	
	2E12 - F064C	Accident Conditions	
	2E12 - F094	Continuous	
k.	2E51 - F086	Accident Conditions	RCIC system
	2E51 - F022	Accident Conditions	
	2E51 - F068	Continuous	
	2E51 - F069	Continuous	
	2E51 - F080	Accident Conditions	
	2E51 - F046	Accident Conditions	
	2E51 - F059	Accident Conditions	
	2E51 - F063	Accident Conditions	
	2E51 - F019	Accident Conditions	
	2E51 - F031	Continuous	
	2E51 - F045	Accident Conditions	
	2E51 - F008	Accident Conditions	
	2E51 - F010	Accident Conditions	
	2E51 - F013	Accident Conditions	
	2E51 - F064	Accident Conditions	
	2E51 - F076	Accident Conditions	
l.	DELETED		
m.	2E22 - F004	Accident Conditions	HPCS system
	2E22 - F012	Accident Conditions	
	2E22 - F015	Continuous	
	2E22 - F023	Accident Conditions	

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A.1

ITS 3.3.8.2

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

LCO 3.3.8.2 3.8.3.4 Two RPS electric power monitoring assemblies for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times. L.1

ACTION:

ACTION A a. With one RPS electric power monitoring assembly for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring assembly to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service. A.2

ACTION B b. With both RPS electric power monitoring assemblies for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring assembly to OPERABLE status within ~~30 minutes~~ 1 hour L.2 or remove the associated RPS MG set or alternate power supply from service. A.2

Add Proposed ACTION C A.3

Add Proposed ACTIONS D, E, and F L.3

SURVEILLANCE REQUIREMENTS

4.8.3.4 The above specified RPS electric power monitoring assemblies shall be determined OPERABLE:

SR 3.3.8.2.1 a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.

SR 3.3.8.2.2

SR 3.3.8.2.3 b. At least once per ~~18~~ 24 months by demonstrating the OPERABILITY of LD.1 LE.1

SR 3.3.8.2.2 overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including

SR 3.3.8.2.3 simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following LA.1

~~setpoints~~ Allowable Values A.4

SR 3.3.8.2.2

1. Overvoltage ≤ 132 VAC with time delay set to ≤ 4 seconds
2. Undervoltage ≥ 108 VAC with time delay set to ≤ 4 seconds
3. Underfrequency ≥ 57 Hz with time delay set to ≤ 4 seconds

M.1

LF.1

Page 2 of 2

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

A.1

LIMITING CONDITION FOR OPERATION

See ITS 3.9.2 — A.2

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

Applicability

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

In-vessel fuel movement — A.3

Applicability

LCO 3.9.1

b. ~~CORE ALTERATIONS~~ shall not be performed using equipment associated with a Refuel position interlock unless at least the following Refuel position interlocks are OPERABLE for such equipment.

SR 3.9.1.1

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Service platform hoist fuel-loaded.

A.4

APPLICABILITY: OPERATIONAL CONDITION 5*# — A.7 — A.6

ACTION: A.5

See ITS 3.9.2 — A.2

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

In-vessel fuel movement — A.3

Action A

c. With any of the above required Refuel position equipment interlocks inoperable, suspend ~~CORE ALTERATIONS~~ with equipment associated with the inoperable Refuel position equipment interlock.

L.1 — Add proposed Required Action A.2.1 and A.2.2

A.6

~~* See Special Test Exceptions 3.10.1 and 3.10.8.~~ ## — A.8 — Moved to ITS 3.10.1

~~# The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.~~ — A.7

The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff. — A.8

ITS 3.10.1

A.1

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Service platform hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5*#.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

<See ITS 3.9.1 and 3.9.2>

LCO 3.10.1

---

*See Special Test Exceptions 3.10.1 and 3.10.3.

#The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

##The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods ~~are verified to~~ remain fully inserted ~~by a second licensed operator or other technically qualified member of the unit technical staff.~~ LA.1

In core cells containing one or more fuel assemblies. L.1

add proposed LCO 3.10.1.b L.1

add proposed ACTION and Surveillance Requirements M.1

Page 4 of 4

ITS 3.9.2

3/4.9 REFUELING OPERATIONS

A.1

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

A.2

L.1

3.9.1 ~~The reactor mode switch shall be~~ OPERABLE ~~and~~ locked ~~in the Shutdown or~~ Refuel ~~position.~~ When the reactor mode switch is locked in the Refuel position:

covered by SR 3.9.2.1

Applicability

LCO 3.9.2

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Service platform hoist fuel-loaded.

See ITS 3.9.1 A.3

APPLICABILITY: OPERATIONAL CONDITION 5~~*#~~ A.6

A.4 A.5

ACTION:

L.1

ACTION A — a. With the reactor mode switch not locked in the ~~Shutdown or~~ Refuel position as specified, ~~suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.~~ L.2

L.1

ACTION A — b. With the one-rod-out interlock inoperable, ~~lock the reactor mode switch in the Shutdown position.~~ L.2

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

See ITS 3.9.1 A.3

A.5

Moved To ITS 3.10.1 A.7

~~*See Special Test Exceptions 3.10.1 and 3.10.3.~~ ##

~~#The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.~~ A.6

##The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

Moved to ITS 3.10.1 A.7

A.1

ITS 3.9.1

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and

2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

A.2 — See ITS 3.9.2

L.2

SR 3.9.1.1 — 4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST ~~within 24 hours prior to the start of~~ ~~and~~ at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable. in-vessel movement

A.2

~~4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.~~

L.3

A.1

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

4.9.1.1 The reactor mode switch shall be verified to be locked in the ~~Shutdown or~~ Refuel position as specified: L.1

~~a. Within 2 hours prior to:~~

~~1. Beginning CORE ALTERATIONS, and~~

~~2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.~~

L.3

b. At least once per 12 hours.

SR 3.9.2.2

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST ~~within 24 hours prior to the start of and~~ at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable. L3

See ITS 3.9.1 A.3

~~4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks that is affected shall be demonstrated OPERABLE by peformance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.~~ L.5

Add proposed Note to SR 3.9.2.2 L.4

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.1.2 and Table 3.3.1.2-1 3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE# ~~and inserted to the normal operating level with:~~ A.5 LA.2

~~a. Continuous visual indication in the control room,~~

SR 3.3.1.2.2.b & SR 3.3.1.2.2.c b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

~~c. The "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn## and shutdown margin demonstrations.~~ LA.3

APPLICABILITY: OPERATIONAL CONDITION 5~~, unless the following conditions are met:~~

Note to SR 3.3.1.2.4

a. No more than four (4) fuel assemblies are present in each core quadrant associated with an SRM;

b. While in core, these four fuel assemblies are in locations adjacent to the SRM; and M.4

~~c. In the case of movable detectors, detector location shall be selected such that each group of fuel assemblies is separated by at least two (2) fuel cell locations from any other fuel assemblies.~~ LA.4

ACTION:

ACTION E With the requirements of the above specification except for control rod insertion not satisfied, immediately suspend all operations involving CORE ALTERATIONS and initiate action to insert all insertable control rods in core cells containing one or more fuel assemblies. A.4 L.5 L.7

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

SR 3.3.1.2.1 1. Performance of a CHANNEL CHECK,

~~2. Verifying the detectors are inserted to the normal operating level, and~~ LA.2

SR 3.3.1.2.2.b & SR 3.3.1.2.2.c 3. Note 1 During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant. add proposed SR 3.3.1.2.2.a and Note 2 M.5

Note (c) to Table 3.3.1.2-1 *The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

~~#The normal or emergency power source may be inoperable.~~ A.5

~~##Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ LA.3

Add proposed Note (b) to T3.3.1.2-1 L.8

Page 5 of 6

A.1

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

Add proposed SNR determination and Note M.1

SR 3.3.1.2.5

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. ~~Within 24 hours prior to the start of CORE ALTERATIONS, and~~ L.4
2. At least once per 7 days.

SR 3.3.1.2.4

c. Verifying that the channel count rate is at least 0.7 cps#:

1. ~~Prior to control rod withdrawal,~~ L.4
2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and
3. At least once per 24 hours.

d. ~~Verifying that the RPS circuitry "shorting links" have been removed within 8 hours prior to and at least once per 12 hours during:~~

1. ~~The time any control rod is withdrawn,## or~~
2. ~~Shutdown margin demonstrations.~~

LA.3

Add proposed SR 3.3.1.2.7 M.3

SR 3.3.1.2.4

20:1 M.6

#Provided signal-to-noise ratio is > ~~2~~. Otherwise, 3 cps.

~~##Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ LA.3

ITS 3.9.3

REFUELING OPERATIONS A.1

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

LCO 3.9.3 — 3.9.3 All control rods shall be inserted.* — A.2

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.** — A.3 — L.1

ACTION:

Action A With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock. — A.2

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 — 4.9.3 All control rods shall be verified to be inserted, except as above specified:

- a. ~~Within 2 hours prior to:~~ — L.2
  1. ~~The start of CORE ALTERATIONS.~~
  2. ~~The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.~~ — A.2
- b. At least once per 12 hours.

~~*Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.~~ — A.2

~~**See Special Test Exception 3.10.3.~~ — A.3

Page 2 of 2

Current Specification 3/4.9.4

~~3/4.9.4 DECAY TIME~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.4 The reactor shall be subcritical for at least 24 hours.~~

~~APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.~~

~~ACTION:~~

~~With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.4 The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.~~

LA.1

Page 2 of 2

Current Specification 3/4.9.5

~~3/4.9.5 COMMUNICATIONS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.5 Direct communication shall be maintained between the control room and refueling platform personnel.~~

~~APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.~~

~~ACTION:~~

~~When direct communication between the control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS.~~

R.1

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.5 Direct communication between the control room and refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.~~

Page 2 of 2

Current Specification 3/4.9.6

3/4.9.6 CRANE AND HOIST

LIMITING CONDITION FOR OPERATION

3.9.6 All cranes and hoists used for handling fuel assemblies or control rods within the reactor pressure vessel shall be OPERABLE.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for crane and hoist OPERABILITY not satisfied, suspend use of any inoperable crane or hoist from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6 Each crane or hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff when the load exceeds:
    1. For the fuel hoist:
        a) 1600 +100/-0 pounds with the NF500 mast.
        b) 1200 ±50 pounds with the 762E974 mast.
    2. 1000 ± 50 pounds for the auxiliary hoist.

b. Demonstrating operation of the loaded interlock when the load exceeds:
    1. For the fuel hoist:
        a) 700 +50/-0 pounds with the NF500 mast.
        b) 485 ±50 pounds and 550 ±50 pounds with the 762E974 mast.
    2. 400 ± 50 pounds for the auxiliary hoist.

c. Demonstrating operation of the fuel hoist downtravel stop when downtravel exceeds 54 feet below the platform rails.

d. Demonstrating operation of the fuel hoist and auxiliary hoist up-travel stops when the grapple is lower than or equal to 8 feet below the platform rails.

e. Demonstrating operation of the fuel hoist slack cable cutoff when the hoist is unloaded.

R.1

Page 2 of 2

Current Specification 3/4.9.7

~~3/4.9.7 CRANE TRAVEL~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.7 Loads over the refueling floor, and over the spent fuel storage pool racks when fuel assemblies are in the racks, shall be restricted as follows:~~

a. ~~All movements of a spent fuel shipping cask shall be controlled by the critical "L" path control system of the Reactor Building crane.~~

b. ~~Loads in excess of 1250 pounds shall not travel over the spent fuel storage pool racks.~~

c. ~~One fuel assembly may be moved over the spent fuel storage pool racks provided that it is not raised above 2 foot clearance over the racks.~~

~~**APPLICABILITY:** At all times.~~

~~**ACTION:**~~

~~With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.7 The spent fuel shipping cask critical "L" path control system of the Reactor Building crane shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during spent fuel shipping cask movement over the refueling floor.~~

LA.1

Page 2 of 2

A.1

REFUELING OPERATIONS

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

A.2

LCO 3.9.6 – 3.9.8 At least 22 feet of water shall be maintained over the top of the reactor pressure vessel flange.

New fuel requirements only moved to LCO 3.9.7 — Moved to ITS 3.9.7

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel ~~while in OPERATIONAL CONDITION 5~~ when the fuel assemblies being handled ~~or the fuel assemblies seated within the reactor vessel are irradiated.~~ A.3

New fuel requirements only moved to LCO 3.9.7 — Moved to ITS 3.9.7

A.2

ACTION:

ACTION A – With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel ~~after placing all fuel assemblies~~ and control rods ~~in a safe condition.~~

LA.1

SURVEILLANCE REQUIREMENTS

L.1

SR 3.9.6.1 – 4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of~~ and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

New fuel requirements only moved to LCO 3.9.7 — Moved to ITS 3.9.7

A.2

ITS 3.9.7

REFUELING OPERATIONS A.1

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

23 feet — L.1

LCO 3.9.7 — 3.9.8 At least ~~22 feet~~ of water shall be maintained over the top of ^the reactor pressure vessel ~~flange~~. — L.1 — the irradiated fuel assemblies seated within

L.1 — new

APPLICABILITY: During handling of ^fuel assemblies or control rods within the reactor pressure vessel ~~while in OPERATIONAL CONDITION 5~~ when ~~the fuel assemblies being handled or~~ the fuel assemblies seated within the reactor vessel are irradiated.

A.2

L.1

ACTION:

new

ACTION A — With the requirements of the above specification not satisfied, suspend all operations involving handling of ^fuel assemblies or control rods within the reactor pressure vessel ~~after placing all fuel assemblies and control rods in a safe condition.~~

LA.1

SURVEILLANCE REQUIREMENTS

L.2

SR 3.9.7.1 — 4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of and~~ at least once per 24 hours during handling of ^fuel assemblies or control rods within the reactor pressure vessel.

new — L.1

A.1

ITS 3.7.8

REFUELING OPERATIONS

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

21 ft 4 inches

LCO 3.7.8 3.9.9 At least ~~23 feet~~ of water shall be maintained over the ~~top of active fuel in~~ irradiated fuel assemblies seated in the spent fuel storage pool racks. A3

A.2 APPLICABILITY: ~~Whenever~~ irradiated fuel assemblies ~~are~~ in the spent fuel storage pool. During movement of

During movement of new fuel assemblies in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool.

ACTION:

ACTION A With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area ~~after placing the fuel assemblies~~ and crane load ~~in a safe condition~~. The provisions of Specification 3.0.3 are not applicable.

LA.2

LA.1

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

Page 2 of 2

REFUELING OPERATIONS

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

<See ITS 3.10.4>

LCO 3.10.3 3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core ~~and/or reactor pressure vessel~~ provided that at least the following requirements are satisfied ~~until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.~~ A.2

APPLICABILITY a. The reactor mode switch is OPERABLE and locked ~~in the Shutdown position or~~ in the Refuel position ~~per Table 1.2 and Specification 3.9.1.~~ <See ITS 3.10.4> L.1

~~b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.~~ A.3

add proposed LCO 3.10.3.c.1 L.2

LCO 3.10.3.c.2 c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

1. May be assumed to be the highest worth control rod ~~required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and~~

~~2. Need not be assumed to be immovable or untrippable.~~ A.4

LCO 3.10.3.c.2
LCO 3.10.3.a d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and ~~electrically or hydraulically~~ disarmed. LA.1

LCO 3.10.3.a e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 4 ~~and 5.~~ <See ITS 3.10.4>

~~ACTION:~~ A.5

add proposed ACTIONS Note A.6

ACTIONS A and B With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

add proposed Required Actions A.2.1, A.2.2, and B.2.1 M.1

add proposed Required Action A.1 Notes A.6

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REFUELING OPERATIONS

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

LCO 3.10.4 3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied ~~until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.~~ A.2

- a. ~~The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.~~ L.1
- b. ~~The source range monitors (SRM) are OPERABLE per Specification 3.9.2.~~ A.3
- LCO 3.10.4.c c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;
  - 1. May be assumed to be the highest worth control rod ~~required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and~~ A.4
  - ~~2. Need not be assumed to be immovable or untrippable.~~
- O 3.10.4.d d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed. LCO 3.10.4.a) LA.1
- LCO 3.10.4.a e. All other control rods are inserted. add proposed LCO 3.10.4.c (first part) and LCO 3.10.4.A M.1

APPLICABILITY: OPERATIONAL CONDITIONS ~~4 and~~ 5*. with LCO 3.9.5 not met A.5

See ITS 3.10.3

ACTION:

ACTION A With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements. add proposed Required Action A.2.1 A.6

Page 3 of 4

A.1

ITS 3.10.3

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

SR3.10.3.2
SR3.10.3.3

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that: L.3 A.2 L.1

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1, L.1 SR3.10.3.1 (See ITS 3.10.4)

b. The SRM channels are OPERABLE per Specification 3.9.2 A.3

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c SR3.10.3.1

add proposed SR 3.10.3.2 Note A.6

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed, and LA.1 SR3.10.3.2 SR3.10.3.3

e. All other control rods are inserted. SR3.10.3.3

add proposed SR3.10.3.1 and SR3.10.3.4 L.2

Page 5 of 6

SURVEILLANCE REQUIREMENTS

SR3.10.4.1
SR3.10.4.2

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that: L.2 A.2

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1. L.1 M.1

b. The SRM channels are OPERABLE per Specification 3.9.2. A.3

SR3.10.4.4 c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c

SR3.10.4.2
SR3.10.4.1 d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically/or hydraulically disarmed, and LA.1

SR 3.10.4.1 e. All other control rods are inserted.

add proposed SR3.10.4.3 and SR3.10-4.5 M.1

A.1

REFUELING OPERATIONS

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

LCO 3.10.5 3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core. A.2

LCO 3.10.5 a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below. L.1

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2. A.3

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied. A.4

LCO 3.10.5.b d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

LCO 3.10.5.a e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

add proposed LCO 3.10.5.c M.1

APPLICABILITY: OPERATIONAL CONDITION 5. with LCO 3.9.4 or LCO 3.9.5 not met A.5

ACTION:

Action A With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

add proposed Required Action A.2.1 A.6

ITS 3.10.5

A.1

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

SR3.10.5.1
SR3.10.5.2

4.9.10.2.1 ~~Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and~~ L.2 ~~at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core,~~ A.2 verify that:

a. ~~The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.~~ L1

b. ~~The SRM channels are OPERABLE per Specification 3.9.2.~~ A.3

c. ~~The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.~~ A.4

SR3.10.5.2 d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

SR 3.10.5.1 e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

~~4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.~~ L3

add proposed SR 3.10.5.3 M.1

Page 4 of 4

A.1

REFUELING OPERATIONS

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.8 — 3.9.11.1 At least one shutdown [A.5] cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE~~@~~ and in operation* ~~with at least:~~ — LA.1

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet above the top of the reactor pressure vessel flange.

ACTION:

ACTION A — a. With no RHR shutdown cooling mode loop OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. — A.2

ACTION B — Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours. — A.3 — A.4

ACTION C — b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

Required Action C.1

SR 3.9.8.1 — 4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. — LA.2

LCO 3.9.8 NOTE — *The shutdown cooling pump be removed from operation for up to 2 hours per 8-hour period.

~~@The normal or emergency power source may be inoperable.~~ — A.5

A.1

ITS 3.9.9

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

A.2

LCO 3.9.9 — 3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE## and at least one loop shall be in operation,* ~~with each loop consisting of at least:~~ — LA.1

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet above the top of the reactor pressure vessel flange.

ACTION:

ACTION A — a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

ACTION C — b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

Add proposed ACTION B — M.1

SURVEILLANCE REQUIREMENTS

Required Action C.1

SR 3.9.9.1 — 4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. — LA.2

LCO 3.9.9 NOTE — *The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

~~##The normal or emergency power source may be inoperable for each loop.~~ — A.2

Page 2 of 2

CTS 3/4.10.1

3/4.10 SPECIAL TEST EXCEPTIONS

~~3/4.10.1 PRIMARY CONTAINMENT INTEGRITY~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.~~

~~APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.~~

~~ACTION:~~

~~With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.~~

M.1

Page 2of2

ITS 3.10.6

A.1

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

of LCO 3.1.6 — A.2

LCO 3.10.6 3.10.2 The sequence constraints imposed on control rod groups by the Rod Worth Minimizer (RWM) per Specification 3.1.4.1 may be suspended by means of bypassing the RWM for the following tests, provided that control rod movement prescribed for this testing is verified by a second licensed operator, or other technically qualified member of the unit technical staff who is present at the reactor control console. — A.3

add proposed LCO 3.10.6.a — A.2

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 10% of RATED THERMAL POWER. — M.1

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2. with LCO 3.1.6 not met — A.4

ACTION:

ACTION A With the requirements of the above specification not satisfied, verify that the RWM is OPERABLE per Specification 3.1.4.1. — A.5

SURVEILLANCE REQUIREMENTS

add proposed SR3.10.6.1 Note — A.6

SR3.10.6.1 4.10.2 When the sequence constraints imposed on control rod groups by the RWM are bypassed, verify;

a. DELETED

SR3.10.6.1 b. That movement of control rods from 75% ROD DENSITY to the RWM low power setpoint is limited to the approved control rod withdrawal sequence during scram and friction tests, — M.2

SR3.10.6.1 c. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3, and

SR3.10.6.1 d. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

add proposed SR3.10.6.2 — A.2

Page 2 of 2

A.1

SPECIAL TEST EXCEPTIONS

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

A.2

LCO 3.10.7 3.10.3 The provisions of ~~Specification 3.9.1, Specification 3.9.3 and~~ Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

~~a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.~~ A.3

LCO 3.10.7.b b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

LCO 3.10.7.d c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

add proposed LCO 3.10.7.a and LCO 3.10. .c A.4

LCO 3.10.7.e d. No other CORE ALTERATIONS are in progress.

add proposed LCO 3.10.7.f M.1

APPLICABILITY: OPERATIONAL CONDITION 5, ~~during shutdown margin demonstrations~~.

with the reactor mode switch in the startup/hot standby position A.5

ACTION:

ACTION B With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

add proposed ACTION A A.4

SURVEILLANCE REQUIREMENTS

4.10.3 ~~Within 30 minutes prior to and~~ at least ~~once per 12 hours~~ during the performance of a shutdown margin demonstration, verify that; L.1

~~a. The source range monitors are OPERABLE per Specification 3.9.2.~~ A.3

add proposed SR 3.10.7.2 and SR 3.10.7.3 Note A.6

SR 3.10.7.2 SR 3.10.7.3 b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

SR 3.10.7.4 c. No other CORE ALTERATIONS are in progress.

add proposed SR 3.10.7.1 and SR 3.10.7.5 A.4

add proposed SR 3.10.7.6 M.1

CTS 3/4.10.5

~~3/4.10.5 OXYGEN CONCENTRATION~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.5 The provisions of Specification 3.6.6.2 may be suspended during the performance of the Startup Test Program until 6 months after initial criticality.~~

~~APPLICABILITY: OPERATIONAL CONDITION 1.~~

~~ACTION~~

~~With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.5 The number of months since initial criticality shall be verified to be less than or equal to 6 months at least once per 31 days during the Startup Test Program.~~

M.1

Page 2 of 2

CTS 3/4.10.6

SPECIAL TEST EXCEPTIONS

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

M.1

Page 2 of 2

3/4.11 RADIOACTIVE EFFLUENTS

A.1

ITS 5.5

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

5.5.9.b 3.11.1.1 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to the limits calculated in the ODCM.

Add Proposed ITS 5.5.9 A.8

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA.6

SURVEILLANCE REQUIREMENTS

5.5.9.b 4.11.1.1 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents ~~at least once per 7 days when radioactive materials are being added to the tank.~~

LA.6

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. A.8

page 24 of 29

RADIOACTIVE EFFLUENTS

3/4 11.2 GASEOUS EFFLUENTS A.1 ITS 5.5

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

add proposed ITS 5.5.9 A.8

5.5.9.a 3.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be limited ~~to less than or equal to 4% by volume.~~

~~APPLICABILITY: Whenever the main condenser air ejector system is in operation.~~

~~ACTION:~~

~~a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA.6

SURVEILLANCE REQUIREMENTS

5.5.9.a 4.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits ~~as required by Table 3.3.7.11-1 of Specification 3.3.7.11.~~

LA.6

Page 25 of 29

ITS 3.7.6

RADIOACTIVE EFFLUENTS

A.1

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

after decay of 30 minutes — A.2

LCO 3.7.6 3.11.2.2 The release rate of the sum of the activities from the noble gases measured prior to the holdup line shall be limited to less than or equal to 3.4 x 10^5 microcuries/second.

and — L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3. with any main steam line not isolated and steam jet air ejector (SJAE) in operation

ACTION:

ACTION A — With the release rate of the sum of the activities of the noble gases prior to the holdup line exceeding 3.4 x 10^5 microcuries/second restore the release rate to within its limit within 72 hours or

ACTION B — be in at least STARTUP with the main steam isolation valves closed within the next ~~6~~ 12 hours. — L.2

Add proposed Required Action B.2 — L.1

SURVEILLANCE REQUIREMENTS

Add proposed Required Actions B.3.1 and B.3.2 — L.3

4.11.2.2.1 The radioactivity rate of noble gases prior to the holdup line shall be continuously monitored in accordance with the ODCM. — LA.1

SR 3.7.6.1 4.11.2.2.2 The release rate of the sum of the activities from noble gases prior to the holdup line shall be determined to be within the limits of Specification 3.11.2.2 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken prior to the holdup line. — LA.2

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the off gas pre-treatment Noble Gas Activity Monitor, — LA.2 of greater than or equal to — M.1 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

Add proposed SR 3.7.6.1 Note — L.4

Page 2 of 2

A.1

ITS Chapter 4.0

4.0 5.0 DESIGN FEATURES

4.1 5.1 SITE

4.1.1 EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

A.2

4.1.2 LOW POPULATION ZONE

5.1.2 The low population zone shall be ~~as shown in Figure 5.1.2-1~~ all the land within a circle with its center at the vent stack and a radius of 3.98 miles.

4.1.1 SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1.1-1.

4.1.1 SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1.1-1.

LA.1

~~5.2 CONTAINMENT~~

~~CONFIGURATION~~

~~5.2.1 The primary containment is a steel lined post-tensioned concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined post-stressed concrete vessel in the shape of a truncated cone closed by a steel dome. The drywell is above a cylindrical steel-lined post-stressed concrete suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 229,538 cubic feet. The suppression chamber has an air region of 164,800 to 168,100 cubic feet and a water region of 128,800 to 131,900 cubic feet.~~

~~DESIGN TEMPERATURE AND PRESSURE~~

~~5.2.2 The primary containment is designed and shall be maintained for:~~

- ~~a. Maximum internal pressure: 45 psig.~~
- ~~b. Maximum internal temperature: drywell 340°F.~~
  ~~suppression chamber 275°F.~~
- ~~c. Maximum external pressure: 5 psig.~~
- ~~d. Maximum floor differential pressure: 25 psid, downward.~~
  ~~5 psid, upward.~~

~~SECONDARY CONTAINMENT~~

~~5.2.3 The secondary containment consists of the Reactor Building, the equipment access structure and a portion of the main steam tunnel and has a minimum free volume of 2,875,000 cubic feet.~~

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A.1

ITS Chapter 4.0

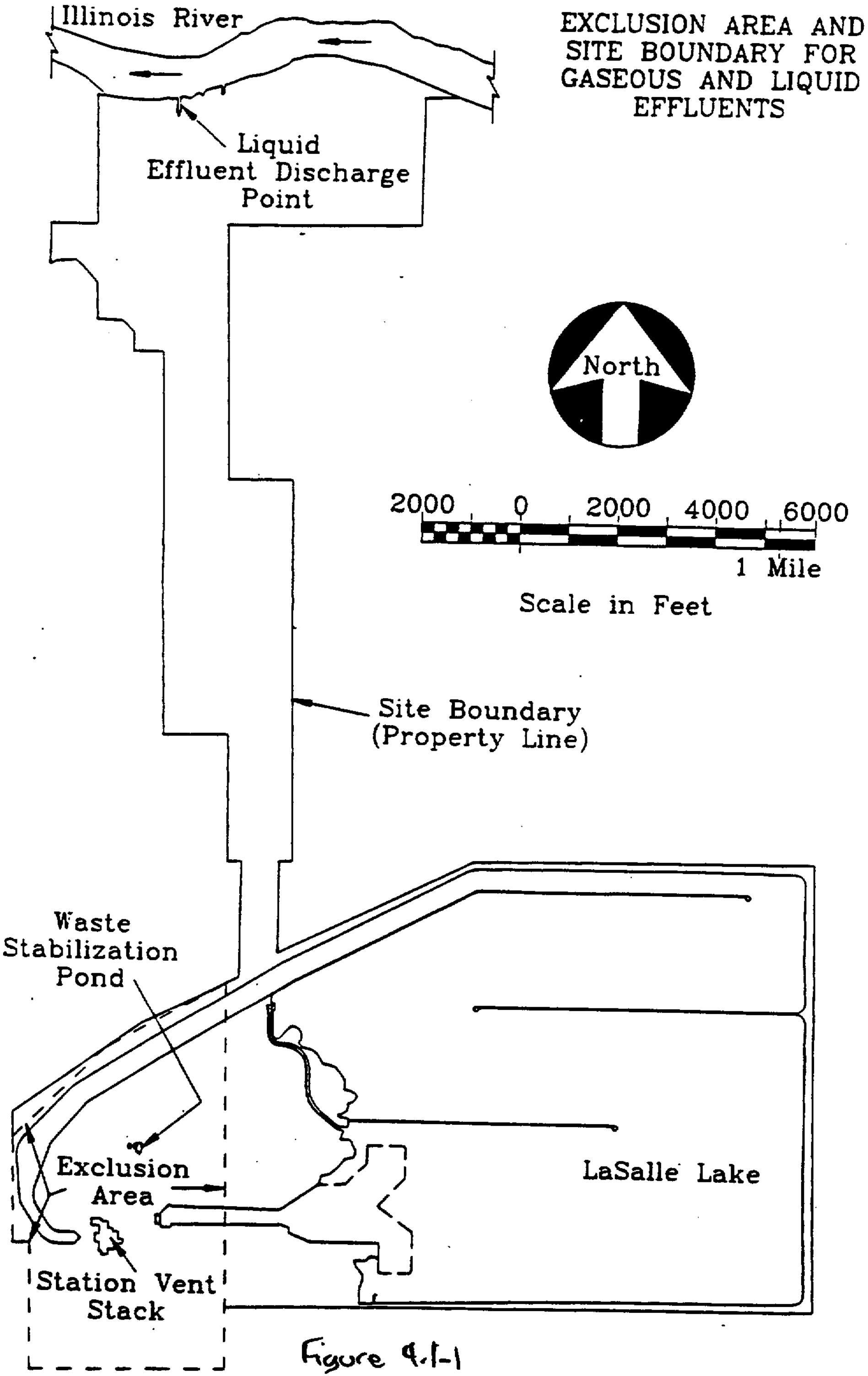
EXCLUSION AREA AND SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS

Figure 4.1-1

Figure 5.1.1-1

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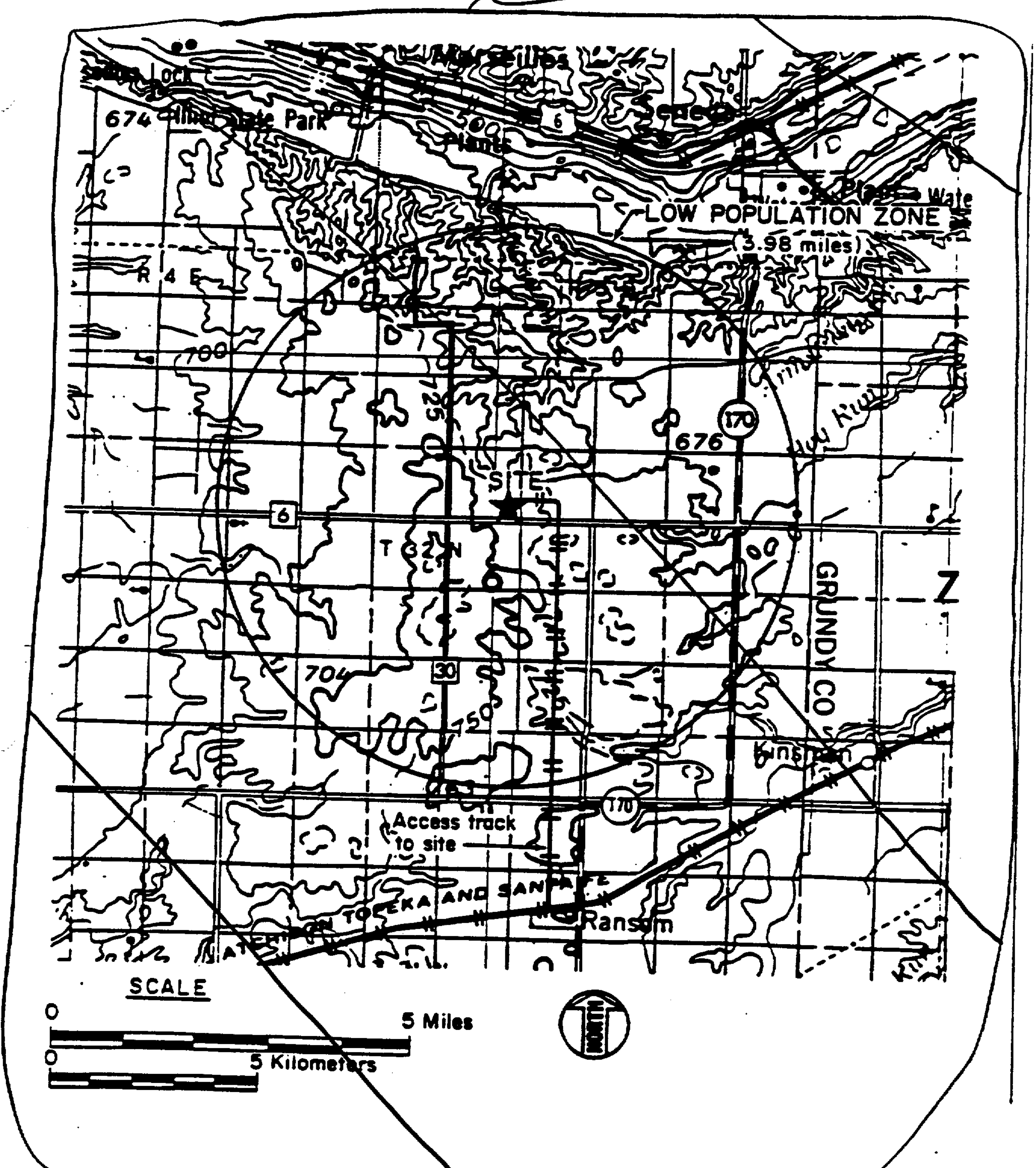
A.1

ITS Chapter 4.0

A.2



LOW POPULATION ZONE

Figure 5.1.2-1

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A.1

ITS Chapter 4.0

4.2 5.3 REACTOR CORE

4.2.1 FUEL ASSEMBLIES

5.3.1 The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy clad A.1 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. The bundles may contain water rods or water boxes. Limited substitutions of Zircalloy or ZIRLO or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. ~~The control rod assembly shall have a nominal axial absorber length of 143 inches.~~ LA.2

LA.1

~~5.4 REACTOR COOLANT SYSTEM~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 The reactor coolant system is designed and shall be maintained:~~

~~a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,~~

~~b. For a pressure of:~~

~~1. 1250 psig on the suction side of the recirculation pumps.~~

~~2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.~~

~~3. 1500 psig from the discharge shutoff valve to the jet pumps.~~

~~c. For a temperature of 575°F.~~

~~VOLUME~~

~~5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.~~

5.5 DELETED

A.1

ITS Chapter 4.0

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY

4.3.1.1 5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to $\leq$ 0.95 when flooded with unborated water, including all calculational uncertainties and biases, as described in Section 9.1 of the FSAR.

b. A nominal 6.26-inch center-to-center distance between fuel assemblies placed in the storage racks.

~~5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooded with water.~~ A.4

4.3.2 DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 feet.

4.3.3 CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4078 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

A.3 moved to ITS Section 5.5

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A.1

ITS 5.5

<See ITS chapter 4.0>

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to ≤ 0.95 when flooded with unborated water, including all calculational uncertainties and biases, as described in Section 9.1 of the FSAR.

b. A nominal 6.26-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooded with water.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4078 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.5.5

5.7.1 The components identified in UFSAR Table ~~5.7.1-1~~ 5.2-4 are designed and shall be maintained within the cyclic or transient limits ~~of Table 5.7.1-1~~.

LA.7

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TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	10,000 power change cycles	75% to 100% to 75% of RATED THERMAL POWER
	80 step change cycles	Loss of Feedwater heaters
	190 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	2000 power change cycles	50% to 100% to 50% of RATED THERMAL POWER
	400 control rod pattern exchanges	Not applicable
	130 hydrostatic pressure tests	Pressurized to ≥ 930 psig and ≤ 1250 psig.

A.1

A.3 moved to ITS Section 5.5

ITS Chapter 4.0

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TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	10,000 power change cycles	75% to 100% to 75% of RATED THERMAL POWER
	80 step change cycles	Loss of Feedwater heaters
	190 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	2000 power change cycles	50% to 100% to 50% of RATED THERMAL POWER
	400 control rod pattern exchanges	Not applicable
	130 hydrostatic pressure tests	Pressurized to ≥ 930 psig and ≤ 1250 psig.

LA.7

ITS 5.5

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5.0 6.0 ADMINISTRATIVE CONTROLS

RESPONSIBILITY

5.1 6.1 ORGANIZATION A.1 ITS 5.1

SEE ITS 5.2

A. Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Manual.

M.1 5.1.1 and shall delegate in writing the succession to this responsibility during his absence

2. The individual filling the ANSI N18.1-1971 Section 4.2.1 position of Plant Manager ("~~Plant~~ Manager"), shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. LA.1

station

SEE ITS 5.2

3. The Chief Nuclear Officer (CNO) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

~~B. The Shift Manager shall be responsible for directing and commanding the overall operation of the facility on his shift. The primary management responsibility of the Shift Manager shall be for safe operation of the nuclear facility on his shift under all conditions.~~ LA.2

SEE ITS 5.2

C. The shift manning for the station shall be as shown in Figure 6.1-3.

page 3 of 4

5.0 6.0 **ADMINISTRATIVE CONTROLS**

5.2 6.1 **ORGANIZATION** A.1 ITS 5.2

5.2.1 A. Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant. LA.1

5.2.1.a 1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Manual.

, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in the Technical Specifications,

5.2.1.b 2. The individual filling the ANSI N18.1-1971 Section 4.2.1 position of Plant Manager ("~~Plant~~ Manager"), shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

Station

A corporate officer LA.1

5.2.1.c 3. ~~The Chief Nuclear Officer (CNO)~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

radiation protection A.2

5.2.1.d 4. The individuals who train the operating staff and those who carry out ~~health physics~~ and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

< SEE ITS 5.1 > B. The Shift Manager shall be responsible for directing and commanding the overall operation of the facility on his shift. The primary management responsibility of the Shift Manager shall be for safe operation of the nuclear facility on his shift under all conditions.

5.2.2 C. The shift manning for the station shall be as shown in Figure 6.1-3.

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A.1

ITS 5.2

5.2.2.b 1. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator who has been designated by the Shift Manager to assume the control room direction responsibility shall be in the Control Room.

5.2.2.d 2. A radiation protection technician* shall be on site when fuel is in the reactor.

~~3. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~ LA.2

4. DELETED

LA.3

~~5. The Independent Safety Engineering Group (ISEG) shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving unit safety. The ISEG shall be composed of at least three, dedicated, full-time engineers of multi-disciplines located on site and shall be augmented on a part-time basis by personnel from other parts of the Commonwealth Edison Company organization to provide expertise not represented in the group. The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification# that these activities are performed correctly and that human errors are reduced as much as practical. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving unit safety to the Manager of Quality and Safety Assessment and the Plant Manager.~~

5.2.2.g 6. The Shift Technical Advisor shall provide advisory technical support to the ~~Shift Manager~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. LA.1

shift manager

5.2.2.d * The radiation protection technician position may be less than the minimum requirement for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required position.

~~# Not responsible for sign-off feature.~~ LA.3

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CTS 6.1.E/F

7. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

8. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

See ITS 5.2

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Health Physics Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives:

1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or

2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection."

See ITS 5.3

~~E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.~~

~~F. Retraining shall be conducted at intervals not exceeding 2 years.~~

LA.1

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A.1

ITS 5.2

5.2.2.e 7. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

5.2.2.f 8. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License. } LA.1

< SEE ITS 5.3 >

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Health Physics Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives:

1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or

2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection."

(See CTS 6.1.E/F)

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.

F. Retraining shall be conducted at intervals not exceeding 2 years.

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A.1

ITS 5.3

See ITS 5.2

7. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

8. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

LA.2

5.3.1 D. Qualifications of the ~~station management and~~ ~~operating~~ unit staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The ~~Health Physics~~ radiation protection Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives:

A.1

LA.1

5.3.1.a 1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or

LA.1

5.3.1.b 2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection."

LA.1

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.

F. Retraining shall be conducted at intervals not exceeding 2 years.

See CTS 6.1.E/F

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A,1

ITS 5.2

~~6. DELETED (The Review and Investigative Function and the Audit Function are described in the Quality Assurance Manual Topical Report CE-1-A).~~

(Next page is 6-13)

A.1

ITS 5.1

FIGURE 6.1-3

MINIMUM SHIFT CREW COMPOSITION(a)(c)

POSITION(b)	MINIMUM CREW NUMBER		
	EACH UNIT IN CONDITION 1, 2, OR 3	ONE UNIT IN CONDITION 1, 2, OR 3, AND ONE UNIT IN CONDITION 4 OR 5 OR DEFUELED	EACH UNIT IN CONDITION 4 OR 5 OR DEFUELED
SM	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA(d)	1	1	None

<SEE ITS 5.2>

(a) This table reflects the total requirements for shift staffing of both units.

With the exception of the Shift Manager, the shift crew composition may be one less than the minimum requirements of Figure 6.1-3 for not more than 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Figure 6.1-3. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(b) Table Notation:

SM Shift Manager with a Senior Reactor Operator license for each unit whose reactor contain: fuel.

SRO Individual with a Senior Reactor Operator license for each unit whose reactor contains fue

During CORE ALTERATIONS on either unit a licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present to observe and directly supervise this operation.

RO An Individual with a Reactor Operator license or a Senior Reactor Operator license for uni assigned. At least one RO shall be assigned to each unit whose reactor contains fuel. Individuals acting as relief operators shall hold a license for both units. Otherwise, for eac unit, provide a relief operator who holds a license for the unit assigned.

AO At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel.

STA Shift Technical Advisor.

5.1.2 (c) While either unit is in CONDITION 1, 2, or 3, an individual with a valid SRO license shall be designated to assume the control room command function. With both Units in CONDITION 4 or 5 or defueled an individual with a valid SRO or RO license shall be designated to assume the control room command function.

A.2

<SEE ITS 5.2> (d) The STA position shall be filled by an individual who meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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A.1 ITS 5.2

FIGURE 6.1-3
MINIMUM SHIFT CREW COMPOSITION(a)(c)

POSITION(b)	MINIMUM CREW NUMBER		
	EACH UNIT IN CONDITION 1, 2, OR 3	ONE UNIT IN CONDITION 1, 2, OR 3, AND ONE UNIT IN CONDITION 4 OR 5 OR DEFUELED	EACH UNIT IN CONDITION 4 OR 5 OR DEFUELED
SM SRO RO	1 1 3	1 1 3	1 None 2
AO	3	3	3
STA(d)	1	1	None

LA.4

5.2.2.a {

LA.4

5.2.2.a (a) This table reflects the total requirements for shift staffing of both units.

5.2.2.c With the exception of the Shift Manager, the shift crew composition may be one less than the minimum requirements of Figure 6.1-3 for not more than 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Figure 6.1-3. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. A.3

(b) Table Notation: LA.4

SM Shift Manager with a Senior Reactor Operator license for each unit whose reactor contains fuel.

SRO Individual with a Senior Reactor Operator license for each unit whose reactor contains fuel

During CORE ALTERATIONS on either unit a licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present to observe and directly supervise this operation. LA.2

5.2.2.b { RO An Individual with a Reactor Operator license or a Senior Reactor Operator license for unit assigned. At least one RO shall be assigned to each unit whose reactor contains fuel. Individuals acting as relief operators shall hold a license for both units. Otherwise, for each unit, provide a relief operator who holds a license for the unit assigned. LA.4

5.2.2.a AO At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel. M.1

STA Shift Technical Advisor. LA.4

<SEE ITS 5.1> (c) While either unit is in CONDITION 1, 2, or 3, an individual with a valid SRO license shall be designated to assume the control room command function. With both Units in CONDITION 4 or 5 an individual with a valid SRO or RO license shall be designated to assume the control room command function.

5.2.2.g (d) The STA position shall be filled by an individual who meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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A.1

ITS 5.7

5.7 6.1.1 HIGH RADIATION AREAS

20.1601 (c)

20.1601

A.2

5.7.1 6.1.1.1 Pursuant to Paragraph ~~20.203(c)(5)~~ of 10 CFR 20, in lieu of the "control device" or "alarm signal" required by paragraph ~~20.203(c)(2)~~ of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr*, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual, i.e., qualified in radiation protection procedures, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~Health Physicist~~ radiation protection manager in the Radiation Work Permit (RWP).

LA.1

5.7.2 6.1.1.2 In addition to the requirements of 6.1.1.1, above, for areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem*, the computer shall be programmed to permit entry through locked doors for any individual requiring access to any such High-High Radiation Areas for the time that access is required.

5.7.3 6.1.1.3 Keys to manually open computer controlled High Radiation Area doors and High-High Radiation Area doors shall be maintained under the Administration control of the shift manager on duty ~~and/~~or the ~~Health Physicist~~.

5.7.4 6.1.1.4 High-High Radiation areas, as defined in 6.1.1.2 above, not equipped with the computerized card readers shall be maintained in accordance with 10 CFR ~~20.203 c.2 (iii)~~, locked except during periods when access to the area is required with positive control over each individual entry, or ~~10 CFR 20.203.c.4.~~ in the case of a High Radiation Area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For

A.2

20.1601 (a)(3) A.2

Page 3 of 4

HIGH RADIATION AREAS (Continued)

<SEE ITS 5.7> individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

5.4 6.2 PLANT OPERATING PROCEDURES AND PROGRAMS <See ITS 5.5>

5.4.1 A. Written procedures shall be established, implemented, and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

5.4.1.b b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

~~c. Station Security Plan implementation,~~

~~d. Generating Station Emergency Response Plan implementation,~~ A.2

~~e. PROCESS CONTROL PROGRAM implementation,~~ LA.1

~~f. OFFSITE DOSE CALCULATION MANUAL implementation, and~~ A.3

5.4.1.c g. Fire Protection Program implementation.

Add proposed TS 5.4.1.d M.1

<SEE ITS 5.7> *Measurement made at 18" from source of radioactivity.

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A.1 ITS 5.7

5.7 **HIGH RADIATION AREAS** (Continued)

5.7.4 individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.2 PLANT OPERATING PROCEDURES AND PROGRAMS

A. Written procedures shall be established, implemented, and maintained covering the activities referenced below:

   a. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

   b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

   c. Station Security Plan implementation,

   d. Generating Station Emergency Response Plan implementation,

   e. PROCESS CONTROL PROGRAM implementation,

   f. OFFSITE DOSE CALCULATION MANUAL implementation, and

   g. Fire Protection Program implementation.

<See ITS 5.4>

~~*Measurement made at 18" from source of radioactivity.~~

A.2

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CTS 6.2.B

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

LA.1

B. ~~Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.~~

C. TECHNICAL REVIEW AND CONTROL

Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Plant Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:

1. Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).

2. Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2, 4.3, 4.4, 4.5.1, or 4.6, and be approved by the Plant Manager.

3. Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation.

4. Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.

5. The Department Head approval authority shall be specified in station procedures.

6. Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.

7. Editorial and Typographical changes shall be made in accordance with station procedures.

<See ITS 5.4>

Page 2 of 2

(See ITS 5.5)

5.4 PLANT OPERATING PROCEDURES AND PROGRAMS (Continued) A.1 ITS 5.4

B. Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

(see CTS 6.2.B)

C. TECHNICAL REVIEW AND CONTROL

LA.2

Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Plant Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:

1. Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).
2. Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2, 4.3, 4.4, 4.5.1, or 4.6, and be approved by the Plant Manager.
3. Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation.
4. Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.
5. The Department Head approval authority shall be specified in station procedures.
6. Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.
7. Editorial and Typographical changes shall be made in accordance with station procedures.

LA.2

D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed and approved in accordance with Specification 6.2.C within 14 days of implementation.

A.4

E. Drills of the emergency procedures described in Specification 6.2.A.d shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

F. The following programs shall be established, implemented, and maintained:

SEE ITS 5.5

1. Primary Coolant Sources Outside Primary Containment

A program to reduce leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

2. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

3. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis,

c. Provisions for maintenance of sampling and analysis equipment.

A.1 ITS 5.5

D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed and approved in accordance with Specification 6.2.C. within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.d shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

SEE ITS 5.4

F. The following programs shall be established, implemented, and maintained:

5.5.2

1. Primary Coolant Sources Outside Primary Containment

A program to reduce leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at ~~refueling cycle~~ 24 month intervals ~~or less~~. LD.1

The provisions of SR 3.0.2 are applicable to the 24 month Frequency for performing integrated system leak tests activities. A.2

~~2. In-Plant Radiation Monitoring~~

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for monitoring, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.~~

LA.1

5.5.3

3. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis,

c. Provisions for maintenance of sampling and analysis equipment.

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5.5.4 4. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

5.5.4.a a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.4.b b. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

5.5.4.c c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

5.5.4.d d. Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.4.e e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

5.5.4.f f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.4.g g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following:

5.5.4.g.1 1. For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

5.5.4.g.2 2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ,

5.5.4.h h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

<See ITS 5.4> PLANT OPERATING PROCEDURES AND PROGRAMS (Continued) A.1 ITS 5.5

5.5.4.i i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.4.k j. Limitations on venting and purging of the containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable,

5.5.4.j k. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

LA.2

~~5. Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~

~~a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,~~

~~b. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~c. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

5.5.6 6. Inservice Inspection Program for Post Tensioning Tendons

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1989, except that the unit 1 and 2 primary containments shall be treated as twin containments even though

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Control Program surveillance frequencies. A.2

⟨See ITS 5.4⟩ A.1 ITS 5.5

5.5.6 the Initial Structural Integrity Tests were not within 2 years of each other.

~~The Onsite Review and Investigative Function shall be responsible for reviewing and approving changes to the Inservice Inspection Program for Post Tensioning Tendons.~~ LA.3

The provisions of 4.0.2 and 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.13 7. Primary Containment Leakage Rate Testing Program

5.5.13.a A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

5.5.13.b The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.6 psig.

5.5.13.c The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.635% of primary containment air weight per day.

5.5.13.d Leakage rate acceptance criteria are:

5.5.13.d.1 a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. 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 tests, and $\leq 0.75\ L_a$ for Type A tests.

5.5.13.d.2 b. Air lock testing acceptance criteria are:

5.5.13.d.2(a) 1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

5.5.13.d.2(b) 2) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

~~The provisions of specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.~~ A.3

5.5.13.e The provisions of specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.8 8. Ventilation Filter Testing Program (VFTP) A.13

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in Regulatory Guide 1.52, Revision 2, dated March 1978~~, and in accordance with ASME N510-1989. A.4 LD.2 LD.3

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the VFTP test frequencies.

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A.1 CTS 5.5

See ITS 5.4

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

5.5.8.a a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05 % when tested in accordance with ASME N510-1989, at the system flowrate specified below:

A.12

CRAF System EMUs

ESF Ventilation System	Flowrate (cfm)
SBGT System	≥ 3600 and ≤ 4400
~~CREF System~~	≥ 3600 and ≤ 4400

5.5.8.b b. Demonstrate for each of the ESF system filter units that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below, when tested in accordance with ASME N510-1989, at the system flowrate specified below:

ESF Ventilation System	Penetration and System Bypass	Flowrate (cfm)
SBGT System	0.05 %	≥ 3600 and ≤ 4400
~~CREF System~~	0.05 %	≥ 3600 and ≤ 4400
CRRF ~~System~~	2.0 %	≥ 18000 and ≤ 28900
AEERRF ~~System~~	2.0 %	≥ 14000 and ≤ 22800

A.12

5.5.8.c c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C, a relative humidity of 70 % and a face velocity as specified below.

ESF Ventilation System	Penetration	Face Velocity (fpm)
SBGT System	0.5 %	40
~~CREF System~~	2.5 %	40
CRRF ~~System~~	15.0 %	80
AEERRF ~~System~~	15.0 %	80

A.12

5.5.8.d d. Demonstrate for each of the ESF systems that the pressure drop across the combined moisture separator, heater, prefilter, HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SBGT System	8	≥ 3600 and ≤ 4400
~~CREF System~~	8	≥ 3600 and ≤ 4400
CRRF ~~System~~	3.0	≥ 18000 and ≤ 28900
AEERRF ~~System~~	3.0	≥ 14000 and ≤ 22800

A.12

CTS 6.3

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

SEE ITS 5.5

e. Demonstrate that the heaters for each of the ESF systems dissipate the electrical power specified below when tested in accordance with ASME N510-1989. These readings shall include appropriate corrections for variations from 480 Volts at the bus.

ESF Ventilation System	Wattage (kw)
SBGT System	≥ 21 and ≤ 25
CREF System	≥ 18 and ≤ 22

~~6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION~~

A.1

~~The following actions shall be taken for REPORTABLE EVENTS:~~

a. ~~The Commission shall be notified and a Licensee Event Report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and~~

b. ~~Each REPORTABLE EVENT shall be reviewed by the Onsite Review and Investigative Function.~~

LA.1

page 2of 2

<See ITS 5.4> PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

5.5.8.e e. Demonstrate that the heaters for each of the ESF systems dissipate the electrical power specified below when tested in accordance with ASME N510-1989. These readings shall include appropriate corrections for variations from 480 Volts at the bus.

A.12 CRAF EMUs

ESF Ventilation System	Wattage (kw)
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~~CREF~~ System	≥ 18 and ≤ 22

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b. Each REPORTABLE EVENT shall be reviewed by the Onsite Review and Investigative Function.

<SEE CTS 6.3>

Add proposed ITS 5.5.11 — M.1

Add proposed ITS 5.5.12 — M.1

CTS 6.4

6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

A.1

If a safety limit is exceeded, the reactor shall be shut down immediately pursuant to Specification 2.1.1, 2.1.2 and 2.1.3, and critical reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Site Vice President or his designated alternate. The incident shall be reviewed by the Onsite and Offsite Review and Investigative Functions and a separate Licensee Event Report for each occurrence shall be prepared in accordance with Section 50.73 to 10 CFR Part 50. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Site Vice President and the Director of Safety Review shall be notified within 24 hours.

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6.5 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years:

1. Records of normal plant operation, including power levels and periods of operation at each power level;
2. Records of principal maintenance and activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety;
3. Records and reports of reportable events;
4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;
5. Records of changes to operating procedures;
6. Shift Manager logs; and
7. Byproduct material inventory records and source leak test results.

<SEE CTS6.5>

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety;
2. Changes made to the plant as it is described in the SAR;
3. Records of new and spent fuel inventory and assembly histories;
4. Updated, corrected, and as-built drawings of the plant;
5. Records of plant radiation and contamination surveys;
6. Records of offsite environmental monitoring surveys;

Page 2 of 2

CTS 6.5

6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

SEE CTS 6.4

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4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;
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5. Records of plant radiation and contamination surveys;
6. Records of offsite environmental monitoring surveys;

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CTS 6.5

PLANT OPERATING RECORDS (Continued)

7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;

LA.1

8. Records of radioactivity in liquid and gaseous wastes released to the environment;

9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);

10. Records of individual staff members indicating qualifications, experience, training, and retraining;

11. Inservice inspections of the reactor coolant system;

12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;

13. Records of reactor tests and experiments;

14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;

15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

16. Records of the service lives of all hydraulic and mechanical snubbers required by specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;

17. Records of analyses required by the radiological environmental monitoring program;

18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM; and

19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted

<SEE ITS 5.6>

Page 4 of 4

A.1

ITS 5.6

PLANT OPERATING RECORDS (Continued)

7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;

8. Records of radioactivity in liquid and gaseous wastes released to the environment;

9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);

10. Records of individual staff members indicating qualifications, experience, training, and retraining;

SEE CTS 6.5

11. Inservice inspections of the reactor coolant system;

12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;

13. Records of reactor tests and experiments;

14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;

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19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

5.6 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted

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A.1 ITS 5.6

~~to the director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.~~ A.2

in accordance with 10CFR 50.4 A.3

A. Routine Reports A.3

LA.1

~~1. Startup Report~~

~~A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.~~

A3

2. Annual Report Add proposed ITS 5.6.1 Note A.4

L.1

5.6.1 A tabulation shall be submitted on an annual basis ~~prior to March 1~~ by April 30 of each year of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions (Note: this tabulation supplements the requirements of Section 20.407 of 10 CFR 20), e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on electronic or pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

A.5 2206

A.5

A.6

~~The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5 shall be included in the Annual Report along with the following information: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours~~

CTS 6.9

prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

3. Annual Radiological Environmental Operating Report*

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

<See ITS 5.6>

4. Annual Radioactive Effluent Release Report**

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety/relief valves, shall be submitted on a monthly basis to the addressees specified in 10 CFR 50.4 no later than the 15th of each month following the calendar month covered by the report.

~~A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.~~ L.1

6. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

---

* A single submittal may be made for a multi-unit station.

** A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

<See ITS 5.6>

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ITS 5.6

~~prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ A.6

5.6.2 3. Annual Radiological Environmental Operating Report*

by 15 L.1

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

5.6.3 4. Annual Radioactive Effluent Release Report**

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in accordance with 10 CFR 50.36a A.7

5.6.4 5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety/relief valves, shall be submitted on a monthly basis to the addressees specified in 10 CFR 50.4 no later than the 15th of each month following the calendar month covered by the report.

A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function. See CTS 6.9

5.6.5 6. Core Operating Limits Report

5.6.5.a a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following: A.4

5.6.2 Note * A single submittal may be made for a multi-unit station. Add 2nd sentence of Proposed ITS 5.6.2 Note

5.6.3 Note ** A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; ~~however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~ A.9

A.1

ITS 5.6

Core Operating Limits Report (Continued)

5.6.5.a.1 (1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

5.6.5.a.2 (2) The minimum Critical Power Ratio (MCPR), scram time dependent MCPR limits, and power and flow dependent MCPR limits for Technical Specification 3.2.3. Effects of analyzed equipment out of service are included. LA1

5.6.5.a.3 (3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

5.6.5.a.4 (4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

5.6.5.b b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. For LaSalle County Station Unit 2, the topical reports are:

5.6.5.b.1 (1) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

5.6.5.b.2 (2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

5.6.5.b.3 (3) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation November 1990.

5.6.5.b.4 (4) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

5.6.5.b.5 (5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.

5.6.5.b.6 (6) Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5.6.5.b.7 (7) Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.

5.6.5.b.8 (8) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

Core Operating Limits Report (Continued)

5.6.5.b.9 (9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

5.6.5.b.10 (10) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.

5.6.5.b.11 (11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

5.6.5.b.12 (12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

5.6.5.b.13 (13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

5.6.5.b.14 (14) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

5.6.5.b.15 (15) Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

5.6.5.b.16 (16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

5.6.5.b.17 (17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

5.6.5.b.18 (18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

5.6.5.b.19 (19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

5.6.5.b.20 (20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

5.6.5.b.21 (21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

Core Operating Limits Report (Continued)

5.6.5.b.22 (22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

5.6.5.b.23 (23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

5.6.5.b.24 (24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

5.6.5.b.25 (25) ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

Core Operating Limits Report (Continued)

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

<SEE ITS 5.6>

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the U.S. Nuclear Regulatory Commission Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Deleted

C. Unique Reporting Requirements

1. Special Reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

LA.1

~~6.7 PROCESS CONTROL PROGRAM (PCP)*~~

~~6.7.1 The PCP shall be approved by the Commission prior to implementation.~~

~~6.7.2 Licensee initiated changes to the PCP:~~

~~a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:~~

~~1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and~~

~~2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~

~~b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.~~

~~*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.~~

Core Operating Limits Report (Continued)

5.6.5.c c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

5.6.5.d d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the U.S. Nuclear Regulatory Commission ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A.2

B. Deleted A.3

C. Unique Reporting Requirements A.8

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2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

SEE CTS 6.7

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*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.

CTS 6.9

6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)*

6.8.1 The ODCM shall be approved by the Commission prior to implementation.

6.8.2 Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the On-Site Review and Investigative Function and the approval of the Plant Manager on the date specified by the On-Site Review and Investigative Function.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

<SEE ITS 5.5>

~~6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS~~

LA.1

~~6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):~~

~~a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Onsite Review and Investigative Function. The discussion of each change shall contain:~~

L.1

~~1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;~~

~~*The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.~~

<SEE ITS 5.5>

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A.1

ITS 5.5

5.5.1 **6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)***

~~6.8.1 The ODCM shall be approved by the Commission prior to implementation.~~ A.9

5.5.1.c 6.8.2 Licensee initiated changes to the ODCM:

5.5.1.c.1 a. Shall be documented and records of reviews performed shall be retained ~~as required by Specification 6.5.B.18~~. This documentation shall contain: A.10

5.5.1.c.1(a) 1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

5.5.1.c.1(b) 2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.~~106~~ 1302 A.9, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.c.2 b. Shall become effective after ~~review and acceptance by the On-Site Review and Investigative Function and~~ LA.3 the approval of the ~~Plant~~ Station LA.8 Manager LA8 ~~on the date specified by the On-Site Review and Investigative Function.~~ LA.3

5.5.1.c.3 c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

<SEE CTS 6.9>

6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Onsite Review and Investigative Function. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

5.5.1 *The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.

CTS 6.9

~~MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Continued)~~

LA.1

2. ~~Sufficient detailed information to totally support the reason for the change without benefit or additional or supplemental information;~~

3. ~~A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;~~

4. ~~An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;~~

5. ~~An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;~~

6. ~~A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period to when the changes are to be made;~~

7. ~~An estimate of the exposure to plant operating personnel as a result of the change; and~~

8. ~~Documentation of the fact that the change was reviewed and found acceptable by the Onsite Review and Investigative Function.~~

b. ~~Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.~~

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