UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 20, 2008

Mr. Barry S. Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT FOR THE CONVERSION TO THE IMPROVED TECHNICAL SPECIFICATIONS WITH BEYOND SCOPE ISSUES (TAC NOS. MD6319-MD6322, MD6324-MD6333, MD6398-MD6403, MD6644-MD6649, AND MD6684)

Dear Mr. Allen:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 279 to Facility Operating License No. NPF-3 for Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The amendment consists of changes to the technical specifications (TSs) and the license conditions for DBNPS in response to your application dated August 3, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072200448), as supplemented by letters dated May 16, 2008 (2 letters) (ADAMS Accession Nos. ML081480464 and ML081430105), July 23, 2008 (ADAMS Accession No. ML082070079), August 7, 2008 (ADAMS Accession No. ML082270658), August 26, 2008 (ADAMS Accession No. ML082600594), and September 3, 2008 (ADAMS Accession No. ML082490154).

The amendment converts the current TSs (CTSs) to the improved TSs (ITSs) and relocates certain requirements to other licensee-controlled documents. The ITSs are based on NUREG-1430, "Standard Technical Specifications (STS) Babcock and Wilcox Plants," Rev. 3.0; "NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," dated July 22, 1993 (58 FR 39132); and 10 CFR 50.36, "Technical Specifications." Technical Specification Task Force changes were also incorporated. The purpose of the conversion is to provide clearer and more readily understandable requirements in the TSs for DBNPS to ensure safe operation. In addition, the amendment includes a number of issues that were considered beyond the scope of NUREG-1430.

Included in the amendments are the following two conditions for the DBNPS operating license: (1) the requirement to relocate certain CTSs requirements into licensee-controlled documents during the implementation of the ITSs, and (2) the schedule for the first performance of new and revised surveillance requirements for the ITSs. These license conditions, which are discussed in the enclosed safety evaluation (SE), are part of the implementation of the ITSs and constitute regulatory commitments that the NRC staff is relying upon in approving the amendment.

The ITSs will become the governing TSs for DBNPS upon the date of implementation. This means that until the implementation of the ITSs is complete, the CTSs shall remain in effect. Upon complete implementation of the ITSs, please submit a letter stating as such within 14 days of the date of completion.

A copy of the related SE is enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions concerning this letter and the SE, contact me at 301-415-3719 or email Cameron.Goodwin@nrc.gov.

Sincerely,

Cameron S. Goodwin, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 279 to NPF-3
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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FIRSTENERGY NUCLEAR OPERATING COMPANY

AND

FIRSTENERGY NUCLEAR GENERATION CORP.

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 279
License No. NPF-3

1. The Nuclear Regulatory Commission (the Commission) has found that:

    A. The application for amendment filed by FirstEnergy Nuclear Operating Company et al. (the licensee), dated August 3, 2007, as supplemented by letters dated May 16, 2008 (2 letters), July 23, 2008, August 7, 2008, August 26, 2008, September 3, 2008, and October 21, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

    C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 279, are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications.

3. This amendment authorizes the relocation of certain Technical Specification requirements and operating license conditions to other licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the other documents, as described in (1) Sections D and E of the NRC staff's Safety Evaluation, and (2) Table LA of Removed Details and Table R of Relocated Specifications attached to the NRC staff's Safety Evaluation, which is enclosed with this amendment.

4. License condition, 2.C.(5), is deleted.

5. New license conditions are added to Appendix C to address performance of new and revised Surveillance Requirements (SRs):

   For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment.

   For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

   For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

   For SRs that existed prior to this amendment whose intervals of performance are being extended the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.

6. This license amendment is effective as of its date of issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: November 20, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 279

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-3	License NPF-3
Page 4	Page 4
Page 6	Page 6
	Page 6b
TSs	TSs
All pages	All pages

2.C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at steady state reactor core power levels not in excess of 2817 megawatts (thermal). Prior to attaining the power level, Toledo Edison Company shall comply with the conditions identified in Paragraph (3) (o) below and complete the preoperational tests, startup tests and other items identified in Attachment 2 to this license in the sequence specified. Attachment 2 is an integral part of this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 279, are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

(a) FENOC shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

(b) Deleted per Amendment 6

(c) Deleted per Amendment 5

2.C(4) Fire Protection

FENOC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report and as approved in the SERs dated July 26, 1979, and May 30, 1991, subject to the following provision:

FENOC may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(5) Deleted per Amendment No. 279.

(6) Antitrust Conditions

FENOC and FirstEnergy Nuclear Generation Corp. shall comply with the antitrust conditions delineated in Condition 2.E of this license as if named therein. FENOC shall not market or broker power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1. FirstEnergy Nuclear Generation Corp. is responsible and accountable for the actions of FENOC to the extent that said actions affect the marketing or brokering of power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1, and in any way, contravene the antitrust license conditions contained in the license.

2.C(9) Implementation of New and Revised Surveillance Requirements

For SRs that are new in Amendment No. 279, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment.

For SRs that existed prior to Amendment No. 279, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

For SRs that existed prior to Amendment No. 279, that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

For SRs that existed prior to Amendment No. 279, whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.

(10) Removed Details and Requirements Relocated to Other Controlled Documents

License Amendment No. 279 authorizes the relocation of certain technical specifications and operating license conditions, if applicable, to other licensee-controlled documents. Implementation of this amendment shall include relocation of these requirements to the specified documents.

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3.3 INSTRUMENTATION (continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

1.0 USE AND APPLICATION

1.1 Definitions

NOTE

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ALLOWABLE THERMAL POWER	ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.
AXIAL POWER IMBALANCE	AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP.
AXIAL POWER SHAPING RODS (APSRs)	APSRs shall be control components used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.
CHANNEL CALIBRATION	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 all devices in the channel required for channel OPERABILITY and 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.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of

1.1 Definitions

CHANNEL CHECK (continued)	the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps.
CONTROL RODS	CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) 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 those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or those listed in ICRP 30, Supplement to Part 1, page 192-212, table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ 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, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

1.1 Definitions

LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal return flow), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE), b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal return flow) that is not identified LEAKAGE; and c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
NUCLEAR HEAT FLUX HOT CHANNEL FACTOR (F_Q)	F_Q shall be the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions.
NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)	$F^N_{\Delta H}$ shall be the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power.

1.1 Definitions

OPERABLE – OPERABILITY	A system, subsystem, train, 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, normal or emergency electrical power, cooling and seal water, lubrication, 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).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Section 14, "Initial Tests and Operation," of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage of the Power in any Core Quadrant (P_{quad}) to the Average Power of all Quadrants (P_{avg}). $QPT = 100\,[\,(P_{quad} / P_{avg}) - 1\,]$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2817 MWt.

1.1 Definitions

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SAFETY FEATURES ACTUATION SYSTEM (SFAS) RESPONSE TIME	The SFAS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its SFAS actuation setpoint at the channel sensor until the SFAS 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 so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and c. There is no change in APSR position.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, trains, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, trains, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, trains, channels, or other designated components in the associated function.

1.1 Definitions

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM (SFRCS) RESPONSE TIME	The SFRCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its SFRCS actuation setpoint at the channel sensor until the SFRCS equipment is capable of performing its safety function (i.e., valves travel to their required positions, pumps discharge pressures reach their required values, etc.). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 280
4	Hot Shutdown[b]	< 0.99	NA	$280 > T_{avg} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

However, when a <u>subsequent</u> train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the <u>first</u> inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

1.3 Completion Times

DESCRIPTION (continued)

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

1.3 Completion Times

EXAMPLES (continued)

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

1.3 Completion Times

EXAMPLES (continued)

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

1.3 Completion Times

EXAMPLES (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

1.3 Completion Times

EXAMPLES (continued)

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

1.3 Completion Times

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, "Surveillance Requirement (SR) Applicability." The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be preformed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

1.4 Frequency

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered;

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-------------------NOTE------------------- Not required to be performed until 12 hours after ≥ 25% RTP. --	
Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance was not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------------------NOTE------------------------------ Only required to be met in MODE 1. --	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency was not met), SR 3.0.4 would require satisfying the SR.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Only required to be performed in MODE 1. -----	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2 or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance was not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-------------------NOTE------------------- Not required to be met in MODE 3. --	
Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency was not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the protective limit shown in the COLR for the various combinations of three and four reactor coolant pump operation.

2.1.1.2 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the SL shown in Figure 2.1.1-1.

2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SAFETY LIMIT VIOLATIONS

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.1.2 is violated, restore RCS pressure and temperature within limits and be in MODE 3 within 1 hour.

2.2.3 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

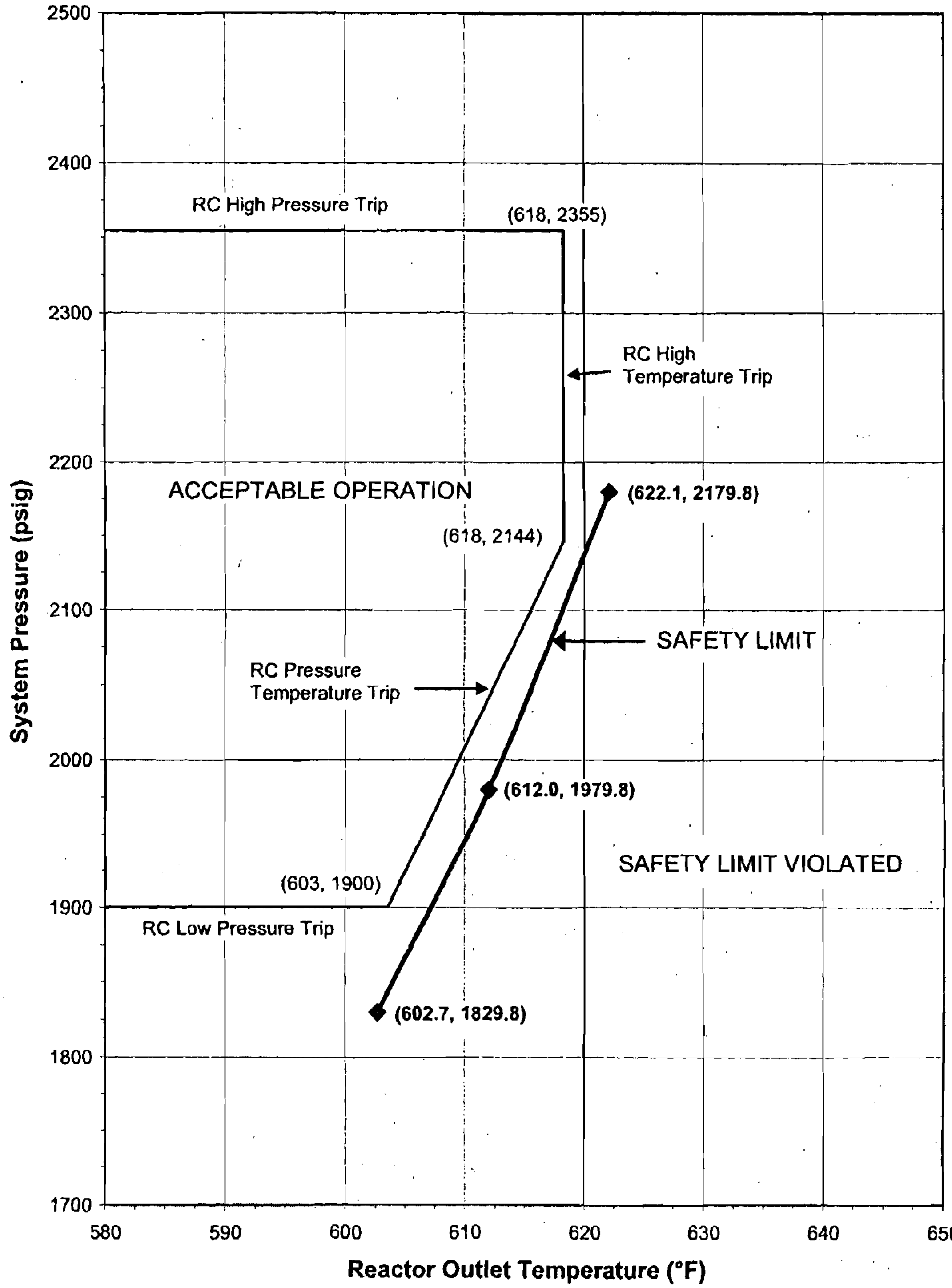

Figure 2.1.1-1 (page 1 of 1)
Reactor Coolant System Departure from Nucleate Boiling Safety Limits

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or

3.0 LCO Applicability

LCO 3.0.4 (continued)

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5	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 LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.
LCO 3.0.6	When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCOs 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 LCO Applicability

LCO 3.0.8

When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

3.0 SR Applicability

SR 3.0.4 (continued)

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is within the limits specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Balance

LCO 3.1.2 The measured core reactivity balance shall be within ± 1% $\Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity balance not within limit.	A.1 Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation. <u>AND</u> A.2 Establish appropriate operating restrictions and SRs.	7 days 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.2.1	-----NOTES----- 1. The predicted reactivity values shall be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. 2. This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify measured core reactivity balance is within ± 1% $\Delta k/k$ of predicted values.	Prior to entering MODE 1 after each fuel loading <u>AND</u> -----NOTE----- Only required after 60 EFPD ---------- 31 EFPD thereafter

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be < 0.9×10^{-4} Δk/k/°F at < 95% RTP and < 0.0×10^{-4} Δk/k/°F at ≥ 95% RTP.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC is within the upper limit specified in the COLR.	Prior to entering MODE 1 after each fuel loading

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	-----NOTE----- If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.3.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. -----	
	Verify extrapolated MTC is within the lower limit specified in the COLR.	Each fuel cycle within 7 EFPDs after reaching an equilibrium boron concentration equivalent to 300 ppm

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 CONTROL ROD Group Alignment Limits

LCO 3.1.4 Each CONTROL ROD shall be OPERABLE.

AND

Each CONTROL ROD shall be aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CONTROL ROD not aligned to within 6.5% of its group average height.	A.1.1 Verify SDM is within limit.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours
	AND	
	A.3 Reduce the High Flux trip setpoint to ≤ 70% of the ALLOWABLE THERMAL POWER.	10 hours
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Verify SDM is within limit.	Once per 12 hours
	AND		
	A.5	Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	72 hours
	AND		
	A.6	-----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----	
		Perform SR 3.2.5.1.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
C. More than one CONTROL ROD not aligned within 6.5% of its group average height.	C.1.1	Verify SDM is within limit.	1 hour
	OR		
	C.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	C.2	Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more CONTROL RODS inoperable.	D.1.1 Verify SDM is within limit. OR D.1.2 Initiate boration to restore SDM to within limit. AND D.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual CONTROL ROD positions are within 6.5% of their group average height.	12 hours
SR 3.1.4.2	Verify CONTROL ROD freedom of movement (trippability) by moving each individual CONTROL ROD that is not fully inserted ≥ 3% in any direction.	92 days
SR 3.1.4.3	------------NOTE------------ With rod drop times determined with less than four reactor coolant pumps operating, operation may proceed provided operation is restricted to the pump combination operating during the rod drop time determination. ---------------------------- Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is ≤ 1.58 seconds from power interruption at the CONTROL ROD drive cabinets to 3/4 insertion (25% withdrawn position) with T_{avg} ≥ 525°F.	Prior to reactor criticality after each removal of the reactor vessel head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod Insertion Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn.

-----NOTE-----
Not required for any safety rod inserted to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One safety rod not fully withdrawn.	A.1.1 Verify SDM is within limit.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Declare the rod misaligned.	1 hour
B. More than one safety rod not fully withdrawn.	B.1.1 Verify SDM is within limit.	1 hour
	<u>OR</u>	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each safety rod is fully withdrawn.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

LCO 3.1.6 Each APSR shall be OPERABLE, unless fully withdrawn, and shall be aligned within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable, not aligned within its limits, or both.	A.1 Perform SR 3.2.3.1.	2 hours <u>AND</u> 2 hours after each APSR movement
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify position of each APSR is within 6.5% of the group average height.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Position Indicator Channels

LCO 3.1.7 The absolute position indicator channel and the relative position indicator channel for each CONTROL ROD and APSR shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each inoperable position indicator channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The relative position indicator channel inoperable for one or more rods.	A.1 Determine the absolute position indicator channel for the rod(s) is OPERABLE.	8 hours <u>AND</u> Once per 8 hours thereafter
B. The absolute position indicator channel inoperable for one or more rods.	B.1.1 Determine position of the rods with inoperable absolute position indicator by actuating the affected rod's zone position reference indicators. <u>AND</u>	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.1.2 Determine rods with inoperable position indicators are maintained at the zone reference indicator position and within the limits specified in LCO 3.1.5, "Safety Rod Insertion Limit," LCO 3.2.1, "Regulating Rod Insertion Limits," or LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," as applicable.	8 hours AND Once per 8 hours thereafter
	OR	
	B.2.1 Place the control groups with nonindicating rods under manual control.	8 hours
	AND	
	B.2.2 Determine the position of the nonindicating rods indirectly with fixed incore instrumentation.	8 hours AND Once per 8 hours thereafter AND ----------NOTE---------- Not applicable during first 8 hour period ------------------------------ 1 hour after motion of nonindicating rods, which exceeds 11% in one direction since the last determination of the rod's position

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR The absolute position indicator channel and the relative position indicator channel inoperable for one or more rods.	C.1 Declare the rod(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify the absolute position indicator channels and the relative position indicator channels agree within the limit specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 1

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.4,	"CONTROL ROD Group Alignment Limits;"
LCO 3.1.5,	"Safety Rod Insertion Limits;"
LCO 3.1.6,	"AXIAL POWER SHAPING ROD (APSR) Alignment Limits;"
LCO 3.2.1,	"Regulating Rod Insertion Limits," for the restricted operation region only;
LCO 3.2.2,	"AXIAL POWER SHAPING ROD (APSR) Insertion Limits;"
LCO 3.2.3,	"AXIAL POWER IMBALANCE Operating Limits;" and
LCO 3.2.4,	"QUADRANT POWER TILT (QPT)"

may be suspended, provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. High Flux trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP;

c. --------------------------------NOTE--------------------------------
Only required when THERMAL POWER is > 20% RTP.

F_Q and $F^N_{\Delta H}$ are maintained within the limits specified in the COLR; and

d. SDM is within the limits specified in the COLR.

APPLICABILITY: MODE 1 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER > 85% RTP. OR High Flux trip setpoint > 10% higher than PHYSICS TESTS power level. OR High Flux trip setpoint > 90% RTP. OR ----------NOTE---------- Only required when THERMAL POWER is > 20% RTP. ------------------------------ F_Q or $F^N_{\Delta H}$ not within limits.	B.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.8.1	Verify THERMAL POWER is ≤ 85% RTP.	1 hour
SR 3.1.8.2	-----NOTE----- Only required to be met when THERMAL POWER is > 20% RTP. ----- Perform SR 3.2.5.1.	2 hours
SR 3.1.8.3	Verify High Flux trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	8 hours
SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.9 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC);"
LCO 3.1.4, "CONTROL ROD Group Alignment Limits;"
LCO 3.1.5, "Safety Rod Insertion Limits;"
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits;"
LCO 3.2.1, "Regulating Rod Insertion Limits," for the sequence and overlap limits, and the insertion limits for the restricted operation region only;
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits;" and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended provided that:

a. THERMAL POWER is ≤ 5% RTP;

b. Reactor trip setpoints on the OPERABLE High Flux channels are set to ≤ 25% RTP;

c. Nuclear instrumentation high startup rate control rod withdrawal inhibit is OPERABLE;

d. SDM is within the limits specified in the COLR; and

e. RCS lowest loop average temperature is ≥ 520°F.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open control rod drive trip breakers.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. SDM not within limit.	B.1 Initiate boration to restore SDM to within limit. <u>AND</u> B.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
C. RCS lowest loop average temperature not within limit.	C.1 Suspend PHYSICS TESTS exceptions.	30 minutes
D. High Flux trip setpoint is not within limit. <u>OR</u> Nuclear instrumentation high startup rate control rod withdrawal inhibit inoperable.	D.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Perform a CHANNEL FUNCTIONAL TEST on each nuclear instrumentation high startup rate control rod withdrawal inhibit and High Flux channel.	Once within 24 hours prior to initiating PHYSICS TESTS
SR 3.1.9.2	Verify the RCS lowest loop average temperature is $\geq$ 520°F.	30 minutes
SR 3.1.9.3	Verify THERMAL POWER is $\leq$ 5% RTP.	1 hour

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.9.4	Verify SDM is within the limits specified in the COLR.	24 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Regulating Rod Insertion Limits

LCO 3.2.1 Regulating rod groups shall be within the physical insertion, sequence, and overlap limits specified in the COLR.

------------------------------------NOTE------------------------------------
Not required for any regulating rod repositioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating rod groups inserted in restricted operation region.	A.1 -------------NOTE------------ Only required when THERMAL POWER is > 20% RTP. ------------------------------- Perform SR 3.2.5.1.	Once per 2 hours
	<u>AND</u>	
	A.2 Restore regulating rod groups to within limits.	24 hours from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group insertion limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Regulating rod groups sequence or overlap limits not met.	C.1 ---NOTE--- Only required when THERMAL POWER is > 20% RTP. --- Perform SR 3.2.5.1.	2 hours
	AND	
	C.2 Restore regulating rod groups to within limits.	4 hours
D. Regulating rod groups inserted in unacceptable operation region.	D.1 Initiate boration to restore SDM to within the limit.	15 minutes
	AND	
	D.2.1 Restore regulating rod groups to within restricted operation region.	2 hours
	OR	
	D.2.2 Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the restricted operation region of the regulating rod group insertion limits.	2 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	12 hours
SR 3.2.1.2	Verify regulating rod groups meet the insertion limits as specified in the COLR.	12 hours
SR 3.2.1.3	Verify SDM is within the limit specified in the COLR.	Within 4 hours prior to achieving criticality

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

LCO 3.2.2 APSRs shall be positioned within the limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. APSRs not within limits.	A.1 -------------NOTE------------ Only required when THERMAL POWER is > 20% RTP. ------------------------------------ Perform SR 3.2.5.1.	Once per 2 hours
	AND	
	A.2 Restore APSRs to within limits.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify APSRs are within acceptable limits specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.3 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AXIAL POWER IMBALANCE not within limits.	A.1 Perform SR 3.2.5.1. <u>AND</u> A.2 Reduce AXIAL POWER IMBALANCE within limits.	Once per 2 hours 24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 40% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT (QPT)

LCO 3.2.4 QPT shall be maintained less than or equal to the steady state limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPT greater than the steady state limit and less than or equal to the transient limit.	A.1.1 Perform SR 3.2.5.1.	Once per 2 hours
	<u>OR</u>	
	A.1.2.1 Reduce THERMAL POWER ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	2 hours <u>OR</u> 2 hours after last performance of SR 3.2.5.1
	<u>AND</u>	
	A.1.2.2 Reduce High Flux trip setpoint and Flux-ΔFlux-Flow trip setpoint ≥ 2% RTP for each 1% of QPT greater than the steady state limit.	10 hours
	<u>AND</u>	
	A.2 Restore QPT to less than or equal to the steady state limit.	24 hours from discovery of failure to meet the LCO

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. QPT greater than the transient limit and less than or equal to the maximum limit due to misalignment of a CONTROL ROD or an APSR.	B.1	Reduce THERMAL POWER ≥ 2% RTP from ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	30 minutes
	AND		
	B.2	Restore QPT to less than or equal to the transient limit.	2 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR QPT greater than the transient limit and less than or equal to the maximum limit due to causes other than the misalignment of either CONTROL ROD or APSR.	C.1	Reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER.	2 hours
	AND		
	C.2	Reduce High Flux trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER.	10 hours
D. Required Action and associated Completion Time for Condition C not met. OR QPT greater than the maximum limit.	D.1	Reduce THERMAL POWER to ≤ 20% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify QPT is within limits as specified in the COLR.	7 days <u>AND</u> -------NOTE------- Only required to be performed if both Condition C was entered and THERMAL POWER is ≥ 60% of ALLOWABLE THERMAL POWER ------------------- When QPT has been restored to less than or equal to the steady state limit, once every hour for 12 hours, or until verified acceptable at ≥ 95% RTP

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Power Peaking Factors

LCO 3.2.5 F_Q and $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F_Q not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% that F_Q exceeds limit.	15 minutes
	AND	
	A.2 Reduce High Flux trip setpoint and Flux-ΔFlux-Flow trip setpoint ≥ 1% RTP for each 1% that F_Q exceeds limit.	10 hours
	AND	
	A.3 Restore F_Q to within limit.	24 hours
B. $F^N_{\Delta H}$ not within limit.	B.1 Reduce THERMAL POWER ≥ RH(%) RTP (specified in the COLR) for each 1% that $F^N_{\Delta H}$ exceeds limit.	15 minutes
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Reduce High Flux trip setpoint and Flux-ΔFlux-Flow trip setpoint ≥ RH(%) RTP (specified in the COLR) for each 1% that $F^N_{\Delta H}$ exceeds limit.	10 hours
	AND	
	B.3 Restore $F^N_{\Delta H}$ to within limit.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER ≤ 20% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.5.1	-----NOTE----- Only required to be performed when specified in LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 1," or when complying with Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," or LCO 3.2.4, "QUADRANT POWER TILT (QPT)." -----	
	Verify F_Q and $F^N_{\Delta H}$ are within limits by using the Incore Detector System to obtain a power distribution map.	As specified by the applicable LCO(s)

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 Four channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

AND

The ultrasonic flow meter (UFM) instrumentation shall be used to perform SR 3.3.1.2 when THERMAL POWER is > 50% RTP.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place channel in bypass or trip.	1 hour
B. Two channels inoperable.	B.1 Place one channel in trip. AND B.2 Place second channel in bypass.	1 hour 1 hour
C. Required Action and associated Completion Time of Condition A or B not met. OR Three or more channels inoperable.	C.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	-------------NOTE------------ Only applicable to Functions 1.a, 3, and 6. -------------------------------- Open all CONTROL ROD drive (CRD) trip breakers.	6 hours
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1	Open all CRD trip breakers.	6 hours
F. UFM instrumentation not used to perform SR 3.3.1.2 when THERMAL POWER is > 50% RTP.	F.1	-------------NOTE------------ Only required if four reactor coolant pumps (RCPs) are operating. -------------------------------- Initiate action to reduce THERMAL POWER to ≤ 98.4% RTP.	Immediately
	AND		
	F.2	-------------NOTE------------ Only required if three RCPs are operating. -------------------------------- Initiate action to reduce THERMAL POWER to ≤ 73.8% RTP.	Immediately
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. (continued)	F.3	------NOTE------ Only required if four RCPs are operating. ------------ Reset High Flux – High Setpoint Allowable Value to ≤ 103.3% RTP	 10 hours

SURVEILLANCE REQUIREMENTS

------NOTE------
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	------NOTES------ 1. Adjust power range channel output if the calorimetric heat balance calculation results exceed power range channel output by > 2% RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. ------------ Compare result of calorimetric heat balance calculation to power range channel output.	 24 hours

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.3	-----NOTES----- 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. For Function 8, flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	 23 days on a STAGGERED TEST BASIS
SR 3.3.1.4	-----NOTES----- 1. Adjust the power range channel imbalance output if the absolute value of the offset error is ≥ 2.5%. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. ---------- Compare results of out of core measured AXIAL POWER IMBALANCE (API_0) to incore measured AXIAL POWER IMBALANCE (API_1) as follows: $(RTP/TP)(API_0 - API_1)$ = offset error.	 31 days
SR 3.3.1.5	Perform CHANNEL FUNCTIONAL TEST.	46 days on a STAGGERED TEST BASIS
SR 3.3.1.6	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.7	-----NOTE----- For Function 8, flow rate measurement sensors are only required to be calibrated. ---------- Perform CHANNEL CALIBRATION.	 24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	---NOTE--- Neutron detectors are excluded from RPS RESPONSE TIME testing. ------	
	Verify that RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 2)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. High Flux -				
a. High Setpoint	1,2[(a)],3[(b)]	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3[(c)(d)] SR 3.3.1.8	≤ 104.9% RTP[(e)] with four pumps operating, and ≤ 80.6% RTP when reset for three pumps operating per LCO 3.4.4, "RCS Loops - MODES 1 and 2"
b. Low Setpoint	2[(f)],3[(f)],4[(f)], 5[(f)]	E	SR 3.3.1.1 SR 3.3.1.3	≤ 5% RTP
2. RC High Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.7	≤ 618°F
3. RC High Pressure	1,2[(a)],3[(b)]	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.8	≤ 2355 psig
4. RC Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.8	≥ 1900 psig

(a) When not in shutdown bypass operation.

(b) With any CRD trip breaker in the closed position, the CRD System capable of rod withdrawal, and not in shutdown bypass operation.

(c) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Technical Requirements Manual.

(e) ≤ 103.3% RTP when reset per ACTION F due to UFM instrumentation not being used to perform SR 3.3.1.2 when THERMAL POWER is > 50% RTP.

(f) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

Table 3.3.1-1 (page 2 of 2)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RC Pressure - Temperature	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.5[(c)(d)] SR 3.3.1.7[(c)(d)]	≥ (16.25 · T_{out} – 7899.0) psig
6. Containment High Pressure	1,2,3[(g)]	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	≤ 4 psig
7. High Flux/Number of Reactor Coolant Pumps On	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.8	≤ 55.1% RTP with one pump operating in each loop, ≤ 0.0% RTP with two pumps operating in one loop and no pumps operating in the other loop, ≤ 0.0% RTP with one pump or no pumps operating
8. Flux - ΔFlux - Flow	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.7 SR 3.3.1.8	Flux – ΔFlux – Flow Allowable Value envelope in COLR
9. Shutdown Bypass High Pressure	2[(f)],3[(f)],4[(f)], 5[(f)]	E	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.7	≤ 1820 psig

(a) When not in shutdown bypass operation.

(c) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Technical Requirements Manual.

(f) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(g) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

3.3 INSTRUMENTATION

3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

LCO 3.3.2 Two RPS Manual Reactor Trip channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Manual Reactor Trip channel inoperable.	A.1	Restore Manual Reactor Trip channel to OPERABLE status.	48 hours
B. Two Manual Reactor Trip channels inoperable.	B.1	Restore one Manual Reactor Trip channel to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Open all CRD trip breakers.	6 hours
D. Required Action and associated Completion Time of Condition A or B not met in MODE 4 or 5.	D.1	Open all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL FUNCTIONAL TEST.	Once prior to each reactor startup if not performed within the previous 7 days

3.3 INSTRUMENTATION

3.3.3 Reactor Protection System (RPS) - Reactor Trip Module (RTM)

LCO 3.3.3 Four RTMs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RTM inoperable.	A.1.1 Trip the associated CRD trip breaker.	1 hour
	<u>OR</u>	
	A.1.2 Remove power from the associated CRD trip breaker.	1 hour
	<u>AND</u>	
	A.2 Physically remove the inoperable RTM.	1 hour
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3. <u>OR</u> Two or more RTMs inoperable in MODE 1, 2, or 3.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2.1 Open all CRD trip breakers.	6 hours
	<u>OR</u>	
	B.2.2 Remove power from all CRD trip breakers.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 4 or 5. OR Two or more RTMs inoperable in MODE 4 or 5.	C.1 OR C.2	Open all CRD trip breakers. Remove power from all CRD trip breakers.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL FUNCTIONAL TEST.	23 days on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.4 CONTROL ROD Drive (CRD) Trip Devices

LCO 3.3.4 The following CRD trip devices shall be OPERABLE:

a. Four CRD trip breakers; and

b. Two silicon controlled rectifier (SCR) relay trip channels.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each CRD trip device.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more CRD trip breakers undervoltage or shunt trip Functions inoperable.	A.1	Trip the associated CRD trip breaker(s).	48 hours
	<u>OR</u>		
	A.2	Remove power from the associated CRD trip breaker(s).	48 hours
B. One or more CRD trip breakers inoperable for reasons other than those in Condition A.	B.1	Trip the associated CRD trip breaker(s).	1 hour
	<u>OR</u>		
	B.2	Remove power from the associated CRD trip breaker(s).	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2.1 Open all CRD trip breakers. OR C.2.2 Remove power from all CRD trip breakers.	6 hours 6 hours 6 hours
D. Required Action and associated Completion Time of Condition A or B not met in MODE 4 or 5.	D.1 Open all CRD trip breakers. OR D.2 Remove power from all CRD trip breakers.	6 hours 6 hours
E. ----------NOTE---------- Required Action E.1 shall be completed whenever this Condition is entered. ------------------------------ One or both SCR relay trip channels inoperable.	E.1 Restore the channel(s) to OPERABLE status.	Prior to entering MODE 4, when in MODE 5 for ≥ 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST on CRD trip breakers.	23 days on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2	Perform CHANNEL FUNCTIONAL TEST on SCR relay trip channels.	24 months

3.3 INSTRUMENTATION

3.3.5 Safety Features Actuation System (SFAS) Instrumentation

LCO 3.3.5 Four channels of SFAS instrumentation for each Parameter in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Parameter.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one channel inoperable.	A.1 Place channel in trip.	1 hour
B. Required Action and associated Completion Time of Condition A not met. OR One or more Parameters with two or more channels inoperable.	B.1 Be in MODE 3. AND B.2 -----NOTE----- Only required for Reactor Coolant System (RCS) Pressure - Low channels. ---------- Reduce RCS pressure < 1800 psig. AND	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3 ----------NOTE---------- Only required for RCS Pressure - Low Low channels. ----------------------------	
	Reduce RCS pressure < 660 psig.	36 hours
	AND	
	B.4 ----------NOTE---------- Only required for Containment Pressure - High, Containment Pressure - High High, and Borated Water Storage Tank - Low Low channels. ----------------------------	
	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

----------------------------------NOTE----------------------------------
Refer to Table 3.3.5-1 to determine which SRs apply to each SFAS instrumentation Parameter.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.5.2	-----NOTE----- When an SFAS channel is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided two other channels of the same SFAS instrumentation Parameter are OPERABLE. ----- Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.5.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.5.5	Verify SFAS RESPONSE TIME within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.5-1 (page 1 of 1)
Safety Features Actuation System Instrumentation

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Coolant System Pressure - Low	1, 2, 3[(a)]	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4 SR 3.3.5.5	≥ 1576.2 psig
2. Reactor Coolant System Pressure - Low Low	1, 2, 3[(b)]	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4 SR 3.3.5.5	≥ 441.42 psig
3. Containment Pressure - High	1, 2, 3, 4	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3 SR 3.3.5.5	≤ 19.38 psia
4. Containment Pressure - High High	1, 2, 3, 4	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3 SR 3.3.5.5	≤ 41.65 psia
5. Borated Water Storage Tank Level - Low Low	1, 2, 3, 4	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3	≥ 101.6 and ≤ 115.4 inches of water

(a) With Reactor Coolant System (RCS) pressure ≥ 1800 psig.

(b) With RCS pressure ≥ 660 psig.

3.3 INSTRUMENTATION

3.3.6 Safety Features Actuation System (SFAS) Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the SFAS Functions below shall be OPERABLE:

a. SFAS; and

b. Containment Spray.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when associated engineered safety features equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SFAS Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	24 months

3.3 INSTRUMENTATION

3.3.7 Safety Features Actuation System (SFAS) Automatic Actuation Logic

LCO 3.3.7 All the SFAS automatic actuation output logics shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when associated engineered safety features equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each automatic actuation output logic.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more automatic actuation output logics inoperable.	A.1	Place associated output logic in trip.	1 hour
	<u>OR</u>		
	A.2	Place associated component(s) in engineered safety features configuration.	1 hour
	<u>OR</u>		
	A.3	Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform automatic actuation output logic CHANNEL FUNCTIONAL TEST.	31 days on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.8 Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)

LCO 3.3.8 Two channels of Loss of Voltage Function and two channels of Degraded Voltage Function EDG LOPS instrumentation per bus shall be OPERABLE.

-----NOTE-----
All Degraded Voltage channels may be bypassed for ≤ 1 minute when starting a reactor coolant pump or circulating water pump.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one channel per bus inoperable.	A.1	Place channel in trip.	1 hour
B. One or more Functions with two channels per bus inoperable.	B.1	Restore one channel to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1	Declare associated EDG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When EDG LOPS instrumentation is placed in an inoperable status solely for performance of a Surveillance, entry into associated Conditions and Required Actions may be delayed up to 2 hours, provided the other channel monitoring the Function for the bus is OPERABLE and the two channels monitoring the Function for the other bus are OPERABLE.

	SURVEILLANCE	FREQUENCY
SR 3.3.8.1	-----NOTE----- The as-left instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit. ----- Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.8.2	-----NOTE----- The as-left instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit. ----- Perform CHANNEL CALIBRATION with Allowable Value as follows: a. Degraded Voltage ≥ 3712 volts (dropout) and ≤ 3771 volts (pickup) with a time delay of ≥ 6.4 seconds and ≤ 7.9 seconds; and b. Loss of Voltage ≥ 2071 volts (dropout) and ≤ 2492 volts (pickup) with a time delay of ≥ 0.42 seconds and ≤ 0.58 seconds.	12 months

3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux

LCO 3.3.9 Two source range neutron flux channels shall be OPERABLE.

-------------------------------NOTE-------------------------------
High voltage to detector may be de-energized with neutron flux > 1E-10 amp on intermediate range channels.

APPLICABILITY: MODES 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux channel inoperable with neutron flux ≤ 1E-10 amp on the intermediate range neutron flux channels.	A.1 Restore channel to OPERABLE status.	Prior to increasing neutron flux
B. Two source range neutron flux channels inoperable with neutron flux ≤ 1E-10 amp on the intermediate range neutron flux channels.	B.1 ------------NOTE------------ Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. ---------------------------------	
	Suspend operations involving positive reactivity changes.	Immediately
	<u>AND</u>	
	B.2 Initiate action to insert all CONTROL RODS.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B: (continued)	B.3	Open CONTROL ROD drive trip breakers.	1 hour
	AND		
	B.4	Verify SDM is within the limits specified in the COLR.	1 hour AND Once per 12 hours thereafter
C. One or more source range neutron flux channels inoperable with neutron flux > 1E-10 amp on the intermediate range neutron flux channels.	C.1	Initiate action to restore affected channel(s) to OPERABLE status.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 Two intermediate range neutron flux channels shall be OPERABLE.

APPLICABILITY: MODE 2,
MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Reduce neutron flux to ≤ 1E-10 amp.	2 hours
B. Required Action and associated Completion Time of Condition A not met. OR Two channels inoperable.	B.1 ------------NOTE------------ Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. -------------------------------- Suspend operations involving positive reactivity changes. AND B.2 Open CRD trip breakers.	 Immediately 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.10.2	-------------------------------NOTE------------------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.11 Steam and Feedwater Rupture Control System (SFRCS) Instrumentation

LCO 3.3.11 The SFRCS instrumentation channels for each Function in Table 3.3.11-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.11-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 Place channel in trip.	1 hour
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more Functions with two or more channels inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 -----NOTE----- Only required for Function 1. ---------- Reduce main steam line pressure < 750 psig. <u>AND</u> B.3 -----NOTE----- Only required for Functions 2, 3, and 4. ---------- Be in MODE 4.	6 hours 12 hours 12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.11-1 to determine which SRs shall be performed for each SFRCS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.11.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.11.2	-----NOTE----- When a channel is placed in an inoperable status solely for performance of the CHANNEL FUNCTIONAL TEST, entry into the associated Conditions and Required Actions may be delayed for up to 8 hours provided the channels providing input to the other actuation channel are OPERABLE. ---------- Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.11.3	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.11.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.11.5	-----NOTE----- "N" equals 2 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. ---------- Verify SFRCS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.11-1 (page 1 of 1)
Steam and Feedwater Rupture Control System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Pressure – Low	1,2,3[a]	4 per steam line	SR 3.3.11.1 SR 3.3.11.2[b][c] SR 3.3.11.3[b][c] SR 3.3.11.5	≥ 600.2 psig
2. Feedwater/Steam Generator Differential Pressure – High	1,2,3	4 per feedwater line	SR 3.3.11.1 SR 3.3.11.2[b][c] SR 3.3.11.3[b][c] SR 3.3.11.5	≤ 176.8 psid
3. Steam Generator Level – Low	1,2,3	4 per steam generator (SG)	SR 3.3.11.1 SR 3.3.11.2[b][c] SR 3.3.11.4[b][c] SR 3.3.11.5	≥ 17.3 inches
4. Loss of RCPs	1,2,3	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3 SR 3.3.11.5	≤ 1384.6 amps and ≥ 106.5 amps

(a) With main steam line pressure ≥ 750 psig during a shutdown and with main steam line pressure > 800 psig during a heatup.

(b) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Technical Requirements Manual.

3.3 INSTRUMENTATION

3.3.12 Steam and Feedwater Rupture Control System (SFRCS) Manual Initiation

LCO 3.3.12 One manual initiation push button for each of the following SFRCS Functions shall be OPERABLE:

a. Auxiliary Feedwater Pump Turbine 1 Initiation;

b. Auxiliary Feedwater Pump Turbine 2 Initiation;

c. Auxiliary Feedwater Pump Turbine 1 Initiation and Steam Generator 1 Isolation; and

d. Auxiliary Feedwater Pump Turbine 2 Initiation and Steam Generator 2 Isolation.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SFRCS Functions inoperable.	A.1 Restore SFRCS Function to OPERABLE status.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.12.1	Perform CHANNEL FUNCTIONAL TEST.	24 months

3.3 INSTRUMENTATION

3.3.13 Steam and Feedwater Rupture Control System (SFRCS) Actuation

LCO 3.3.13 Channels 1 and 2 of each Logic Function shown below shall be OPERABLE:

a. Auxiliary Feedwater Initiation;

b. Auxiliary Feedwater and Main Steam Valve Control;

c. Main Steam Line Isolation; and

d. Main Feedwater Isolation.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Logic Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channel 1 Logic Functions inoperable with all channel 2 Logic Functions OPERABLE. <u>OR</u> One or more channel 2 Logic Functions inoperable with all channel 1 Logic Functions OPERABLE.	A.1 Restore inoperable channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	-----NOTE----- When a channel is placed in an inoperable status solely for performance of the CHANNEL FUNCTIONAL TEST, entry into the associated Conditions and Required Actions may be delayed for up to 8 hours provided the other actuation channel is OPERABLE. ----- Perform CHANNEL FUNCTIONAL TEST.	31 days

3.3 INSTRUMENTATION

3.3.14 Fuel Handling Exhaust - High Radiation

LCO 3.3.14 Two channels of Fuel Handling Exhaust - High Radiation shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool building.

ACTIONS

-----NOTE-----

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Declare the associated Spent Fuel Pool Area Emergency Ventilation System train inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.14.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.14.2	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.14.3	Perform CHANNEL CALIBRATION with a trip setpoint of $\leq$ 2 times Background.	18 months

3.3 INSTRUMENTATION

3.3.15 Station Vent Normal Range Radiation Monitoring

LCO 3.3.15 Two channels of station vent normal range radiation monitoring instrumentation shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One channel inoperable.	A.1	Isolate the Control Room Normal Ventilation System.	7 days
	AND		
	A.2	-----------NOTE----------- Only applicable in MODES 1, 2, 3, and 4. ------------------------------	
		Place one OPERABLE Control Room Emergency Ventilation System (CREVS) train in operation.	7 days
B. Two channels inoperable.	B.1	Isolate the Control Room Normal Ventilation System.	1 hour
	AND		
	B.2	-----------NOTE----------- Only applicable in MODES 1, 2, 3, and 4. ------------------------------	
		Place one OPERABLE CREVS train in operation.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.15.2	-----NOTE----- When a channel is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 3 hours provided the other channel is OPERABLE. ---------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.15.3	Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.16 Anticipatory Reactor Trip System (ARTS) Instrumentation

LCO 3.3.16 The ARTS instrumentation channels for each Function in Table 3.3.16-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.16-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1	-----NOTE----- Only applicable for Function 3. -------------- Trip the control rod drive trip breaker associated with the inoperable channel.	1 hour
	AND		
	A.2	-----NOTE----- Only applicable for Functions 1 and 2. -------------- Restore required channel to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1	-----NOTE----- Only applicable for Function 1. --------------- Reduce THERMAL POWER to ≤ 45% RTP.	 6 hours
	AND		
	B.2	-----NOTE----- Only applicable for Functions 2 and 3. --------------- Be in MODE 2.	 6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

Refer to Table 3.3.16-1 to determine which SRs apply to each ARTS instrumentation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.16.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.16.2	Perform CHANNEL FUNCTIONAL TEST.	23 days on a STAGGERED TEST BASIS
SR 3.3.16.3	Perform CHANNEL CALIBRATION.	46 days on a STAGGERED TEST BASIS

Table 3.3.16-1 (page 1 of 1)
Anticipatory Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Turbine Trip	> 45% RTP	3	SR 3.3.16.1 SR 3.3.16.3
2. Trip of Both Main Feed Pump Turbines	1	3	SR 3.3.16.1 SR 3.3.16.3
3. Output Logic	1	4	SR 3.3.16.2

3.3 INSTRUMENTATION

3.3.17 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.17 The PAM instrumentation for each Function in Table 3.3.17-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1	Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	---------------NOTE-------------- Only applicable to Functions other than Functions 13, 14, and 15. --------------------------------------	
		Initiate action in accordance with Specification 5.6.5.	Immediately
	<u>OR</u>		
	B.2	---------------NOTE-------------- Only applicable to Functions 13, 14, and 15. --------------------------------------	
		Enter the Condition referenced in Table 3.3.17-1 for the channel.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with two required channels inoperable.	C.1	Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1	Enter the Condition referenced in Table 3.3.17-1 for the channel.	Immediately
E. As required by Required Action B.2 or D.1 and referenced in Table 3.3.17-1.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 4.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.17-1	F.1	Initiate action in accordance with Specification 5.6.5.	Immediately

SURVEILLANCE REQUIREMENTS

NOTE

These SRs apply to each PAM instrumentation Function in Table 3.3.17-1 except where identified in the SR.

SURVEILLANCE		FREQUENCY
SR 3.3.17.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.17.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION for Functions 1, 11, 12, 14, 15, 16, and 17.	18 months
SR 3.3.17.3	Perform CHANNEL CALIBRATION for Functions 2, 3, 4, 5, 6, 7, 8, 9, 10, and 13.	24 months

Table 3.3.17-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.2 or D.1
1.	Wide Range Neutron Flux	2	E
2.	Reactor Coolant Loop Outlet Temperature	2 per loop	E
3.	Reactor Coolant Loop Pressure	2 per loop	E
4.	Reactor Coolant Hot Leg Level (Wide Range)	2	E
5.	Containment Water Level (Wide Range)	2	E
6.	Containment Pressure (Wide Range)	2	E
7.	Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path[(a)(b)]	E
8.	Containment High Range Radiation	2	F
9.	Pressurizer Level	2	E
10.	Steam Generator Startup Range Level	2 per SG	E
11.	Incore Thermocouples	2 per core quadrant	E
12.	Auxiliary Feedwater Flow Rate	2 per SG	E
13.	Steam Generator Outlet Steam Pressure	1 per SG	E
14.	High Pressure Injection Flow	1 per injection line	E
15.	Low Pressure Injection (Decay Heat Removal) Flow	1 per train	E
16.	Borated Water Storage Tank Level	2	E
17.	Neutron Flux (Source Range)	2	E

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

3.3 INSTRUMENTATION

3.3.18 Remote Shutdown System

LCO 3.3.18 The remote shutdown monitoring instrumentation Functions shall be OPERABLE.

AND

The control circuit and transfer switch Functions required for a serious control room or cable spreading room fire shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met for remote shutdown monitoring instrumentation Functions.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours
C. Required Action and associated Completion Time of Condition A not met for control circuit and transfer switch Functions.	C.1 Initiate action in accordance with Specification 5.6.7.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.18.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.18.2	-----NOTE----- Reactor trip breaker indication and control rod position switches are excluded from this Surveillance. ----- Perform CHANNEL CALIBRATION for each required instrumentation channel.	24 months
SR 3.3.18.3	Verify each control circuit and transfer switch required for a serious control room or cable spreading room fire is capable of performing the intended function.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for loop pressure, hot leg temperature, and RCS total flow rate shall be within the limits specified below:

a. With four reactor coolant pumps (RCPs) operating:

RCS loop pressure shall be ≥ 2064.8 psig, RCS hot leg temperature shall be ≤ 610°F, and RCS total flow rate shall be ≥ 389,500 gpm; and

b. With three RCPs operating:

RCS loop pressure shall be ≥ 2060.8 psig, RCS hot leg temperature shall be ≤ 610°F, and RCS total flow rate shall be ≥ 290,957 gpm.

APPLICABILITY: MODE 1.

-----NOTE-----
RCS loop pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	---NOTE--- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. --- Verify RCS loop pressure ≥ 2064.8 psig with four RCPs operating or ≥ 2060.8 psig with three RCPs operating.	12 hours
SR 3.4.1.2	---NOTE--- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. --- Verify RCS hot leg temperature ≤ 610°F.	12 hours
SR 3.4.1.3	Verify RCS total flow ≥ 389,500 gpm with four RCPs operating or ≥ 290,957 gpm with three RCPs operating.	12 hours
SR 3.4.1.4	---NOTE--- Not required to be performed until 24 hours after stable thermal conditions are established at ≥ 70% RTP. --- Verify RCS total flow rate is within limit by measurement.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 525°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1	Be in MODE 2 with k_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 525°F.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 shall be completed whenever this Condition is entered. ------------------------------ Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. ----------NOTE---------- Required Action C.2 shall be completed whenever this Condition is entered. ------------------------------ Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limit. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.3.1	-----NOTE----- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and:

1. THERMAL POWER is < 80.6% RTP;

2. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 1.a (High Flux - High Setpoint), Allowable Value of Table 3.3.1-1 is reset for three RCPs operating; and

3. LCO 3.3.1, Function 8 (Flux-ΔFlux-Flow), Allowable Value of Table 3.3.1-1 is reset for three RCPs operating.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO 3.4.4.b.2 not met.	A.1 Satisfy the requirements of LCO 3.4.4.b.2.	10 hours
B. Required Action and associated Completion Time of Condition A not met. OR Requirements of LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE and one RCS loop shall be in operation.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RCS loop inoperable.	A.1	Restore RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 4.	12 hours
C. Two RCS loops inoperable. OR Required RCS loop not in operation.	C.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND		
	C.2	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify one RCS loop is in operation.	12 hours
SR 3.4.5.2	Verify, for each required RCS loop, SG secondary side water level is: a. ≥ 18 inches above the lower tube sheet if associated reactor coolant pump is operating; or b. ≥ 35 inches above the lower tube sheet if reactor coolant pumps are not operating.	12 hours
SR 3.4.5.3	------------NOTE------------ Not required to be performed until 24 hours after a required pump is not in operation. ------------------------------ Verify correct breaker alignment and indicated power available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and one loop shall be in operation.

-----NOTE-----
All reactor coolant pumps (RCPs) and DHR pumps may be de-energized for ≤ 1 hour provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required loop inoperable.	A.1	Initiate action to restore a second loop to OPERABLE status.	Immediately
	<u>AND</u>		
	A.2	-----NOTE----- Only required if one DHR loop is OPERABLE. -----	
		Be in MODE 5.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two required loops inoperable. OR Required loop not in operation.	B.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND		
	B.2	Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required DHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify, for each required RCS loop, SG secondary side water level is: a. ≥ 18 inches above the lower tube sheet if associated reactor coolant pump is operating; or b. ≥ 35 inches above the lower tube sheet if reactor coolant pumps are not operating.	12 hours
SR 3.4.6.3	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7

Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and one loop shall be in operation.

-----NOTE-----

The DHR pump of the loop in operation may be removed from operation for ≤ 1 hour provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately
B. Two required loops inoperable. OR Required loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.7.1	Verify required DHR or RCS loop is in operation.	12 hours
SR 3.4.7.2	Verify, for each required RCS loop, SG secondary side water level is ≥ 35 inches above the lower tube sheet.	12 hours
SR 3.4.7.3	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two decay heat removal (DHR) loops shall be OPERABLE and one DHR loop shall be in operation.

-----NOTES-----

1. All DHR pumps may be removed from operation for ≤ 15 minutes when switching from one loop to another provided:

   a. The maximum RCS temperature is ≤ 190°F;

   b. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM);" and

   c. No draining operations to further reduce the RCS water volume are permitted.

2. One DHR loop may be inoperable for ≤ 2 hours for Surveillance testing provided that the other DHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DHR loop inoperable.	A.1 Initiate action to restore DHR loop to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No required DHR loop OPERABLE. OR Required DHR loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND	
	B.2 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.8.1	Verify required DHR loop is in operation.	12 hours
SR 3.4.8.2	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 228 inches; and

b. A minimum of 85 kW of essential pressurizer heaters OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Restore level to within limit.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours
C. Capacity of essential pressurizer heaters less than limit.	C.1 Restore essential pressurizer heater capacity.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level ≤ 228 inches.	12 hours
SR 3.4.9.2	Verify capacity of essential pressurizer heaters is ≥ 85 kW.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings $\leq$ 2525 psig.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Pilot Operated Relief Valve (PORV)

LCO 3.4.11 The PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PORV inoperable.	A.1 Close block valve. AND A.2 Remove power from block valve.	1 hour 1 hour
B. Block valve inoperable.	B.1 Close block valve. AND B.2 Remove power from block valve.	1 hour 1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-----NOTE----- Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. ---------- Perform one complete cycle of the block valve.	92 days
SR 3.4.11.2	Perform one complete cycle of the PORV.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP)

LCO 3.4.12 The Decay Heat Removal (DHR) System relief valve shall be OPERABLE with:

a. A lift setting of ≤ 330 psig; and

b. The Reactor Coolant System (RCS) to DHR System isolation valves open with control power removed.

APPLICABILITY: MODES 4 and 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DHR System relief valve inoperable due to one or more RCS to DHR System isolation valves closed.	A.1 Open RCS to DHR System isolation bypass valves. AND A.2 Verify RCS to DHR System isolation bypass valves open.	1 hour Once per 24 hours
B. DHR System relief valve inoperable due to one or more RCS to DHR System isolation valves with control power not removed.	B.1 Remove control power from RCS to DHR System isolation valves.	1 hour
C. DHR System relief valve inoperable for reasons other than Condition A or B.	C.1 Restore DHR System relief valve to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and Associated Completion Time not met.	D.1	Disable capability of both high pressure injection pumps to inject water into the RCS.	1 hour
	AND		
	D.2	Disable makeup pump suction automatic transfer to the borated water storage tank on low makeup tank level.	8 hours
	AND		
	D.3	Verify makeup tank level ≤ 73 inches.	8 hours
	AND		
	D.4	Verify RCS pressure and pressurizer level in Acceptable Region of Figure 3.4.12-1 or 3.4.12-2, as applicable.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify RCS to DHR isolation valves open with control power removed.	24 hours
SR 3.4.12.2	Verify DHR System relief valve lift setpoint ≤ 330 psig in accordance with the Inservice Testing (IST) Program.	In accordance with the IST Program

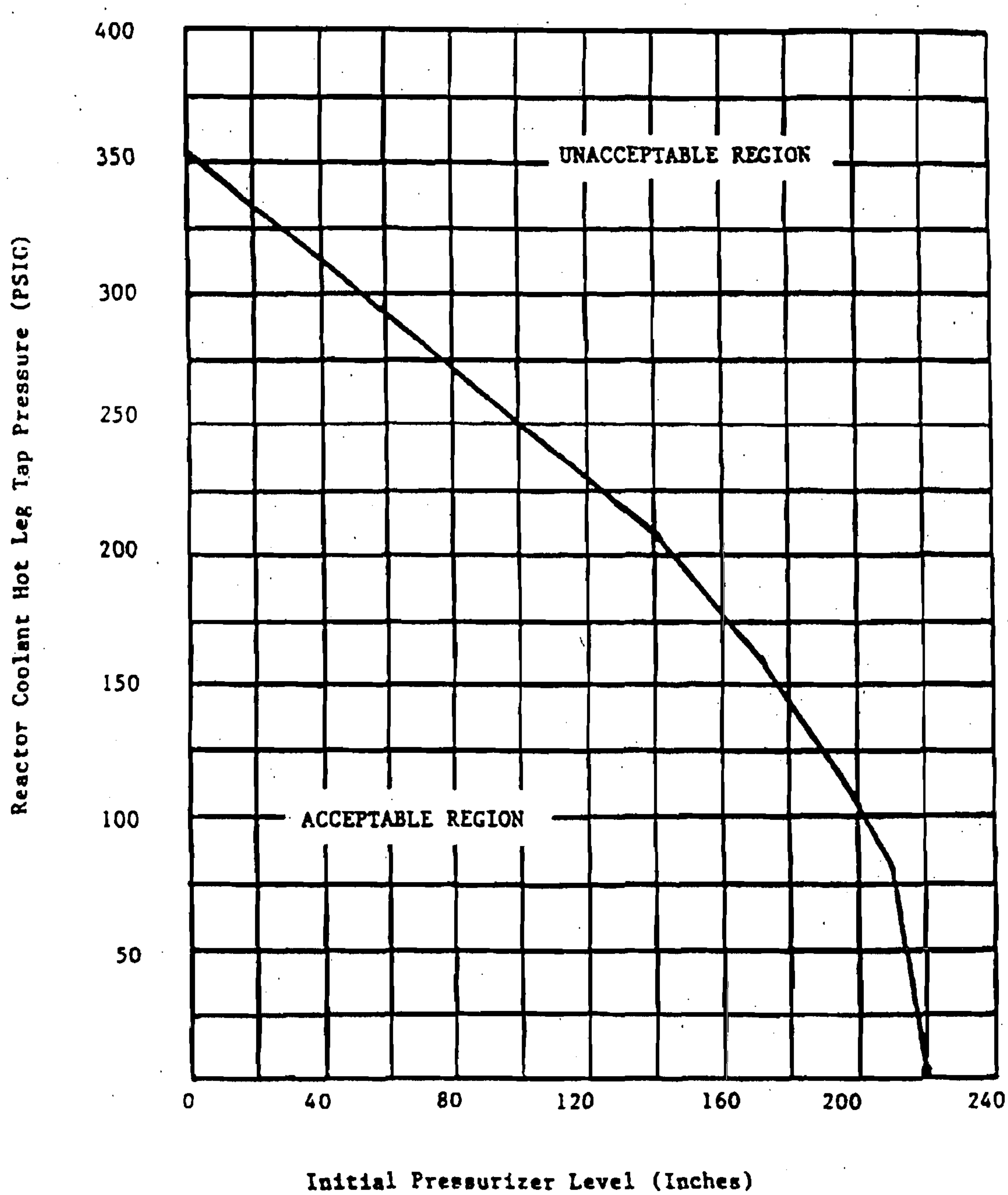

Figure 3.4.12-1
RCS Pressure Versus Pressurizer Level Limit
for Inoperable DHR System Relief Valve in MODE 4

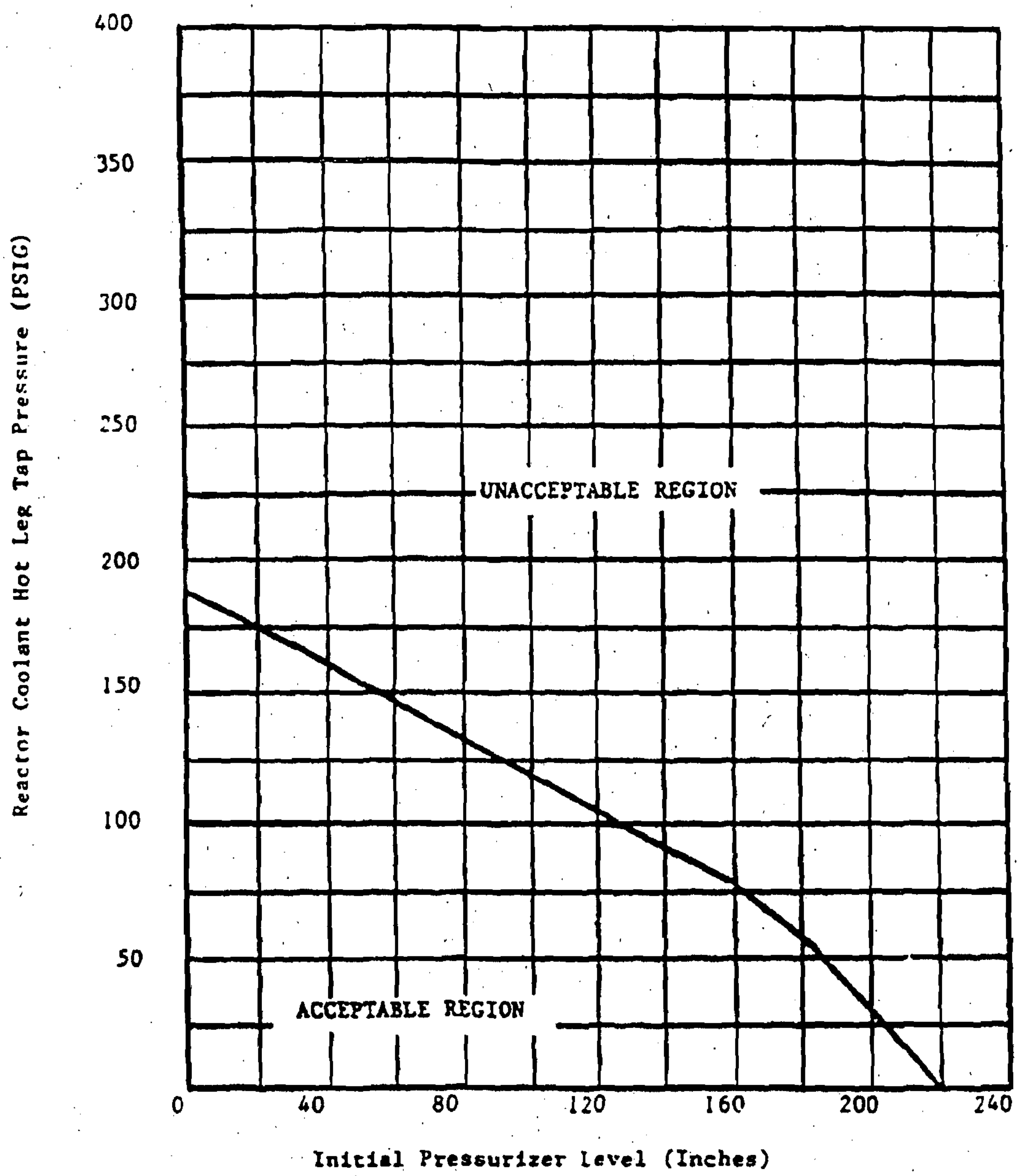

Figure 3.4.12-2
RCS Pressure Versus Pressurizer Level Limit
for Inoperable DHR System Relief Valve in MODE 5 and
MODE 6 when the reactor vessel head is on

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-----NOTES----- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	-----NOTE----- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limits.

<u>AND</u>

The Decay Heat Removal (DHR) System interlock function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except the valves in the DHR flow path when in, or during the transition to or from, the DHR mode of operation and the DHR System interlock function.

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each flow path.
2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	-----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.2.a and be on the RCS pressure boundary or the high pressure portion of the system. ----------	
	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. Decay Heat Removal (DHR) System interlock function inoperable.	C.1	Isolate the affected line by use of two closed deactivated automatic valves.	4 hours
	OR		
	C.2	----------NOTE---------- Only applicable if RCS pressure < 328 psig. ------------------------------	
		Restore the interlock function to OPERABLE status.	Prior to increasing RCS pressure ≥ 328 psig

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	Perform CHANNEL CHECK on the DHR System interlock channel common to Safety Features Actuation System (SFAS) instrumentation.	12 hours
SR 3.4.14.2	-----------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. --- Verify: a. Leakage from each RCS PIV is equivalent to ≤ 5.0 gpm at an RCS pressure of 2155 psig; and b. When current measured rate is > 1 gpm, the current measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and 5.0 gpm by 50%.	24 months <u>AND</u> Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months
SR 3.4.14.3	-----------------------------NOTE----------------------------- Not required to be met when the DHR System interlock function is disabled in accordance with LCO 3.4.12. --- Verify DHR System interlock function prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ 328 psig.	24 months
SR 3.4.14.4	-----------------------------NOTE----------------------------- Not required to be met when the DHR System interlock function is disabled in accordance with LCO 3.4.12. --- Verify DHR System interlock function causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ 328 psig.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.5	Perform CHANNEL CALIBRATION on the DHR System interlock channels.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump monitor; and

b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1 ------------NOTE------------ Not required until 12 hours after establishment of steady state operation. -------------------------------- Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.2 Restore required containment sump monitor to OPERABLE status.	30 days
B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
	OR	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1.2	------------NOTE------------ Not required until 12 hours after establishment of steady state operation. ------------------------------	
		Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	B.2	Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. Both required monitors inoperable.	D.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required containment atmosphere radioactivity monitor.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.15.3	Perform CHANNEL CALIBRATION of required containment atmosphere radioactivity monitor.	18 months
SR 3.4.15.4	Perform CHANNEL CALIBRATION of required containment sump monitor.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 530°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	-----NOTE----- LCO 3.0.4.c is applicable.	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 in unacceptable region of Figure 3.4.16-1. <u>OR</u> Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 530°F.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	7 days
SR 3.4.16.2	-----------------------------NOTE----------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days <u>AND</u> Between 2 and 6 hours after THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period
SR 3.4.16.3	-----------------------------NOTE----------------------------- Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. -- Determine $\bar{E}$.	184 days

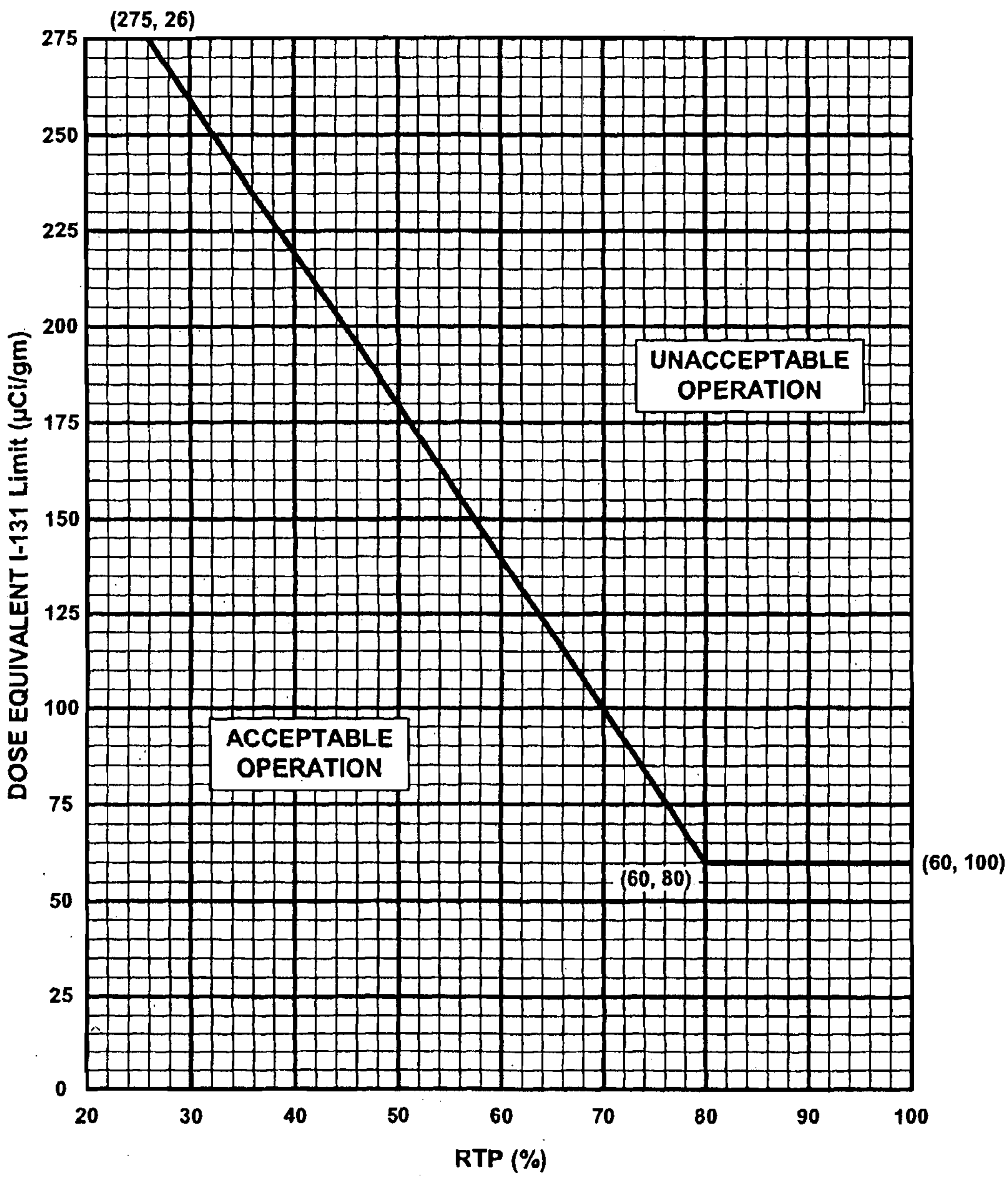

Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit
Versus Percent of RATED THERMAL POWER With Reactor Coolant
Specific Activity > 1.0 µCi/gm DOSE EQUIVALENT I-131

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1	Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND		
	A.2	Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flooding Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce RCS pressure to ≤ 800 psig.	6 hours 18 hours
D. Two CFTs inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each CFT isolation valve is fully open.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.2	Verify borated water volume in each CFT is ≥ 12.6 feet and ≤ 13.3 feet.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is ≥ 580 psig and ≤ 620 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each CFT is ≥ 2600 ppm and ≤ 3500 ppm.	31 days AND --------NOTE-------- Only required to be performed for affected CFT ----------------------- Once within 6 hours after each solution volume increase of ≥ 80 gallons that is not the result of addition from the borated water storage tank
SR 3.5.1.5	Verify power is removed from each CFT isolation valve operator.	31 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

------------------------------NOTE------------------------------
The borated water storage tank (BWST) outlet and containment emergency sump valves may be considered OPERABLE when the associated valve motors are de-energized, provided the valves are not otherwise inoperable.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure injection (LPI) subsystem inoperable.	A.1 Restore LPI subsystem to OPERABLE status.	7 days
B. One or more trains inoperable for reasons other than Condition A.	B.1 Restore train(s) to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours
D. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.3	Verify ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.	24 months AND Prior to declaring ECCS OPERABLE after draining ECCS piping
SR 3.5.2.4	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.5	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.6	Verify the correct position of each mechanical stop for the following valves: a. DH-14A; and b. DH-14B.	24 months

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.2.7	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	24 months
SR 3.5.2.8	Verify the following: a. Each BWST outlet valve and containment emergency sump valve actuate to the correct position on a manual actuation of the containment emergency sump valve; and b. The actuation time of each BWST outlet valve and containment emergency sump valve is $\leq$ 75 seconds.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 One ECCS low pressure injection (LPI) subsystem shall be OPERABLE.

-----NOTE-----
The borated water storage tank (BWST) outlet and containment emergency sump valves may be considered OPERABLE when the associated valve motors are de-energized, provided the valves are not otherwise inoperable.

APPLICABILITY: MODE 4.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to ECCS LPI subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required ECCS LPI subsystem inoperable.	A.1	Initiate action to restore required ECCS LPI subsystem to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	For all equipment required to be OPERABLE, the following SRs are applicable: SR 3.5.2.1 SR 3.5.2.6 SR 3.5.2.2 SR 3.5.2.7 SR 3.5.2.3 SR 3.5.2.8 SR 3.5.2.4 SR 3.5.2.5	In accordance with applicable SRs

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Borated Water Storage Tank (BWST)

LCO 3.5.4 The BWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. BWST boron concentration not within limits. OR BWST water temperature not within limits.	A.1 Restore BWST to OPERABLE status.	8 hours
B. BWST inoperable for reasons other than Condition A.	B.1 Restore BWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	-------------------------NOTE------------------------ Only required to be performed when ambient air temperature is < 35°F or > 90°F. --- Verify BWST borated water temperature is ≥ 35°F and ≤ 90°F.	24 hours
SR 3.5.4.2	Verify BWST borated water volume is ≥ 500,100 gallons and ≤ 550,000 gallons.	7 days
SR 3.5.4.3	Verify BWST boron concentration is ≥ 2600 ppm and ≤ 2800 ppm.	7 days

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

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

ACTIONS

-----NOTES-----

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	-----NOTES----- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. -----	
	A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	1 hour

ACTIONS (continued)

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

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-----NOTES----- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

-----NOTES-----

1. Penetration flow paths except for 48 inch containment purge and exhaust valve penetration flow paths may be unisolated intermittently under administrative controls.

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

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

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two or more containment isolation valves. ----- One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D or E.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours

ACTIONS (continued)

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ----------NOTE---------- Only applicable to penetration flow paths with only one containment isolation valve. --------------------------------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	AND		
	C.2	----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. --------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more penetration flow paths with one or more containment purge or exhaust valves not within purge and exhaust valve leakage limits.	D.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
	AND		
	D.2	-----NOTES----- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	AND		
	D.3	Perform SR 3.6.3.5 for the resilient seal containment purge and exhaust valves closed to comply with Required Action D.1.	Once per 92 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Secondary containment bypass leakage not within limit.	E.1 Restore secondary containment bypass leakage to within limit.	4 hours
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 48 inch containment purge and exhaust valve is closed with control power removed.	31 days
SR 3.6.3.2	-------------------------NOTE------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.3.3	-----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ----- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.5	Perform leakage rate testing for containment purge and exhaust valves with resilient seals.	Within 72 hours after each valve closure, if valve opened in MODE 1, 2, 3, or 4 <u>AND</u> Prior to entering MODE 4 from MODE 5 if valve opened in other than MODE 1, 2, 3, or 4 <u>AND</u> Prior to entering MODE 2 from MODE 3 each time the plant has been in any combination of MODE 3, 4, 5, or 6 for > 72 hours, if not performed in the previous 184 days
SR 3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.6.3.7	Verify the combined leakage for all secondary containment bypass leakage paths is $\leq 0.03\ L_a$.	In accordance with the Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -14 inches water gauge and ≤ +25 inches water gauge.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Air Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment air cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours
C. One required containment air cooling train inoperable.	C.1 Restore required containment air cooling train to OPERABLE status.	7 days
D. One containment spray train and one required containment air cooling train inoperable.	D.1 Restore containment spray train to OPERABLE status. OR D.2 Restore required containment air cooling train to OPERABLE status.	72 hours 72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two required containment air cooling trains inoperable.	E.1	Restore one required containment air cooling train to OPERABLE status.	72 hours
F. Required Action and associated Completion Time of Condition C, D, or E not met.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2	Be in MODE 5.	36 hours
G. Two containment spray trains inoperable. OR Any combination of three or more required trains inoperable.	G.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.2	Operate each required containment air cooling train for $\geq$ 15 minutes.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.3	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.4	Verify each required containment air cooling train starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.6.5	Verify each required containment air cooling train cooling water flow rate is ≥ 1150 gpm.	24 months
SR 3.6.6.6	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.7	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.8	Verify each spray nozzle is unobstructed.	10 years

3.6 CONTAINMENT SYSTEMS

3.6.7 Trisodium Phosphate Dodecahydrate (TSP) Storage

LCO 3.6.7 The TSP storage baskets shall contain ≥ 290 ft^3 of TSP.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TSP storage baskets contain < 290 ft^3 of TSP.	A.1 Restore TSP storage baskets to ≥ 290 ft^3 of TSP.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify contained volume of TSP in the TSP storage baskets is within limit.	24 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSSVs inoperable.	A.1 Reduce power to less than the reduced power requirement of Equation 3.7.1-1.	4 hours
	AND	
	A.2 Reduce the High Flux trip setpoint in accordance with Equation 3.7.1-1.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more steam generators with less than two MSSVs OPERABLE. OR One or more steam generators with no MSSVs with a lift setting of 1050 psig ± 3% OPERABLE.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-----NOTE----- Only required to be performed in MODES 1 and 2. --------------- Verify each MSSV lift setpoint per Table 3.7.1-1 in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
Main Steam Safety Valve Lift Settings

NUMBER OF VALVES	LIFT SETTING (psig ± 3%)
2 MSSVs/steam generator	1050
7 MSSVs/steam generator	1100

$$\frac{WY}{Z} = SP; \quad RP = \frac{Y}{Z} \times 100\%$$

W = High Flux trip setpoint for four pump operation as specified in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

Y = Total OPERABLE MSSV relieving capacity per steam generator based on summation of individual OPERABLE MSSV relief capacities per steam generator lb/hour.

Z = Required relieving capacity per steam generator of 6,585,600 lb/hour.

SP = High Flux trip setpoint (not to exceed W).

RP = Reduced power requirement (not to exceed RTP).

Equation 3.7.1-1 (page 1 of 1)
Reduced Power and High Flux Trip Setpoint
Versus OPERABLE Main Steam Safety Valves

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1	Restore MSIV to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 2.	6 hours
C. ----------NOTE---------- Separate Condition entry is allowed for each MSIV. ------------------------------ One or more MSIVs inoperable in MODE 2 or 3.	C.1	Close MSIV.	8 hours
	<u>AND</u>		
	C.2	Verify MSIV is closed.	Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify isolation time of each MSIV is within limits.	In accordance with the Inservice Testing Program
SR 3.7.2.2	Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Stop Valves (MFSVs), Main Feedwater Control Valves (MFCVs), and associated Startup Feedwater Control Valves (SFCVs)

LCO 3.7.3 Two MFSVs, MFCVs, and associated SFCVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when all MFSVs, MFCVs, and associated SFCVs are closed or isolated by a closed manual valve.

ACTIONS

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

Separate Condition entry is allowed for each MFSV, MFCV, and SFCV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFSVs inoperable.	A.1 Close or isolate MFSV.	72 hours
	<u>AND</u>	
	A.2 Verify MFSV is closed or isolated.	Once per 7 days
B. One or more MFCVs inoperable.	B.1 Close or isolate MFCV.	72 hours
	<u>AND</u>	
	B.2 Verify MFCV is closed or isolated.	Once per 7 days
C. One or more SFCVs inoperable.	C.1 Close or isolate SFCV.	72 hours
	<u>AND</u>	
	C.2 Verify SFCV is closed or isolated.	Once per 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two valves in the same flow path inoperable.	D.1	Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time of each MFSV is within limits.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify the isolation time of each MFCV and SFCV is within limits.	24 months
SR 3.7.3.3	Verify each MFSV, MFCV, and SFCV actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Turbine Stop Valves (TSVs)

LCO 3.7.4 Four TSVs shall be OPERABLE.

APPLICABILITY: MODES 1,
MODES 2 and 3 except when all TSVs are closed.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each TSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more TSVs inoperable.	A.1 Close inoperable TSV.	8 hours
	AND	
	A.2 Verify inoperable TSV is closed	Once per 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify isolation time of each TSV is within limits.	24 months
SR 3.7.4.2	Verify each TSV actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW)

LCO 3.7.5 Three EFW trains shall be OPERABLE, consisting of:

a. Two Auxiliary Feedwater (AFW) trains; and

b. The Motor Driven Feedwater Pump (MDFP) train.

-------------------------------------NOTE-------------------------------------
Only the MDFP train is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

-------------------------------------NOTE-------------------------------------
LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AFW train inoperable due to one inoperable steam supply. OR ----------NOTE---------- Only applicable if MODE 2 has not been entered following refueling. --------------------------- One AFW train inoperable in MODE 3 following refueling.	A.1 Restore AFW train to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One EFW train inoperable for reasons other than Condition A in MODE 1, 2, or 3.	B.1	Restore EFW train to OPERABLE status.	72 hours
C. One AFW train inoperable due to one inoperable steam supply.	C.1	Restore the steam supply to the AFW train to OPERABLE status.	48 hours
<u>AND</u>	<u>OR</u>		
MDFP train inoperable.	C.2	Restore the MDFP train to OPERABLE status.	48 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
<u>OR</u>	<u>AND</u>		
Two EFW trains inoperable for reasons other than Condition C in MODE 1, 2, or 3.	D.2	Be in MODE 4.	12 hours
E. Three EFW trains inoperable in MODE 1, 2, or 3.	E.1	----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ------------------------------	
		Initiate action to restore one EFW train to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required MDFP train inoperable in MODE 4.	F.1	Initiate action to restore MDFP train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	----------NOTE---------- In MODE 1 ≤ 40% RTP and MODES 2, 3, and 4, the MDFP train valves are allowed to be in the non-correct position, provided the valves are capable of being locally realigned to the correct position. ------------------------------ Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the AFW pumps, that is not locked, sealed, or otherwise secured in position, is in the correct position.	 31 days
SR 3.7.5.2	----------NOTE---------- Not required to be performed until 24 hours after reaching 800 psig in the steam generators. ------------------------------ Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	 92 days
SR 3.7.5.3	----------NOTE---------- Not required to be performed until 73 hours after MDFP train is aligned to the AFW System. ------------------------------ Operate the MDFP train.	 92 days

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.5.4	-----NOTE----- Not required to be performed until 24 hours after reaching 800 psig in the steam generators. ----- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.5.5	-----NOTE----- Not required to be performed until 24 hours after reaching 800 psig in the steam generators. ----- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.7.5.6	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tanks to each steam generator.	Prior to entering MODE 2 following refueling or whenever plant has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days
SR 3.7.5.7	Verify proper alignment of the required MDFP flow paths by verifying flow from the condensate storage tanks to each steam generator.	Prior to entering MODE 3 following refueling or whenever plant has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.8	Perform CHANNEL CHECK on each AFW train Steam Generator Level Control System.	12 hours
SR 3.7.5.9	Perform CHANNEL FUNCTIONAL TEST on each AFW train Steam Generator Level Control System.	31 days
SR 3.7.5.10	Perform CHANNEL CALIBRATION on each AFW train Steam Generator Level Control System.	24 months

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tanks (CSTs)

LCO 3.7.6 The CSTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The CSTs inoperable.	A.1	Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter
	AND		
	A.2	Restore CSTs to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4 without reliance on steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify usable volume in the CSTs is $\geq$ 270,300 gal.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW loops shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCW loop inoperable.	A.1 ------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by CCW. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal loop made inoperable by CCW. ------------------------------------- Restore CCW loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----NOTE----- Isolation of CCW flow to individual components does not render CCW System inoperable. -----	
	Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.7.3	Verify each required CCW pump starts automatically on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS loops shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SWS loop inoperable.	A.1	----------NOTES---------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal loop made inoperable by SWS. ------------------------------- Restore SWS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	------------------------NOTE------------------------ Isolation of SWS flow to individual components does not render the SWS inoperable. --- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.8.3	Verify each required SWS pump starts automatically on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of UHS is ≥ 562 ft International Great Lakes Datum.	24 hours
SR 3.7.9.2	Verify average water temperature of UHS is ≤ 90°F.	24 hours

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

-----NOTES-----

1. The control room envelope (CRE) boundary may be opened intermittently under administrative control.

2. Only the CRE boundary is required to be OPERABLE during movement of irradiated fuel assemblies.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable for reasons other than Condition B.	A.1	Restore CREVS train to OPERABLE status.	7 days
B. One or more CREVS trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	B.1	Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>		
	B.2	Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	<u>AND</u>		
	B.3	Restore CRE boundary to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. CRE boundary inoperable during movement of irradiated fuel assemblies.	D.1	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREVS train for ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	Verify Control Room Normal Ventilation System isolates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.5	Verify the system makeup flow rate is ≥ 270 cfm and ≤ 330 cfm when supplying the control room with outside air.	24 months

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.11 Two CREATCS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREATCS train inoperable.	A.1 Restore CREATCS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CREATCS train has the capability to remove the assumed heat load.	24 months

3.7 PLANT SYSTEMS

3.7.12 Station Emergency Ventilation System (EVS)

LCO 3.7.12 Two Station EVS trains shall be OPERABLE.

-----NOTE-----
The shield building area negative pressure boundary may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Station EVS train inoperable.	A.1	Restore Station EVS train to OPERABLE status.	7 days
B. Two Station EVS trains inoperable due to inoperable shield building area negative pressure boundary.	B.1	Restore shield building area negative pressure boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each Station EVS train for ≥ 15 minutes.	31 days
SR 3.7.12.2	Perform required Station EVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each Station EVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.12.4	Verify one Station EVS train can attain a negative pressure ≥ 0.25 inches water gauge in the annulus ≤ 4 seconds after the flow rate is ≥ 7200 cfm and ≤ 8800 cfm.	24 months on a STAGGERED TEST BASIS
SR 3.7.12.5	Verify each Station EVS filter cooling bypass damper can be opened.	24 months

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Area Emergency Ventilation System (EVS)

LCO 3.7.13 Two Spent Fuel Pool Area EVS trains shall be OPERABLE.

-----NOTE-----
The spent fuel pool area negative pressure boundary may be opened under administrative control.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool building.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Spent Fuel Pool Area EVS train inoperable.	A.1	Restore Spent Fuel Pool Area EVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	Place OPERABLE Spent Fuel Pool Area EVS train in operation.	Immediately
	<u>OR</u>		
	B.2	Suspend movement of irradiated fuel assemblies in the spent fuel pool building.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two Spent Fuel Pool Area EVS trains inoperable.	C.1	Suspend movement of irradiated fuel assemblies in the spent fuel pool building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each Spent Fuel Pool Area EVS train for ≥ 15 minutes.	31 days
SR 3.7.13.2	Perform required Spent Fuel Pool Area EVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each Spent Fuel Pool Area EVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.13.4	Verify one Spent Fuel Pool Area EVS train can maintain a negative pressure ≥ 0.125 inches water gauge relative to outside atmosphere.	24 months on a STAGGERED TEST BASIS
SR 3.7.13.5	Verify each Spent Fuel Pool Area EVS filter cooling bypass damper can be opened.	24 months

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Water Level

LCO 3.7.14 The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 ------------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------- Suspend movement of irradiated fuel assemblies in spent fuel pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be ≥ 630 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	-----NOTE----- LCO 3.0.3 is not applicable. ------------------	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>	
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	<u>OR</u>	
	A.2.2 Initiate action to perform a fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Storage

LCO 3.7.16 Fuel assemblies stored in the spent fuel pool shall be placed in the spent fuel pool storage racks in accordance with the criteria shown in Figure 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ------------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------ Initiate action to move the noncomplying fuel assembly to an allowable location.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1.	Prior to storing the fuel assembly in the spent fuel pool

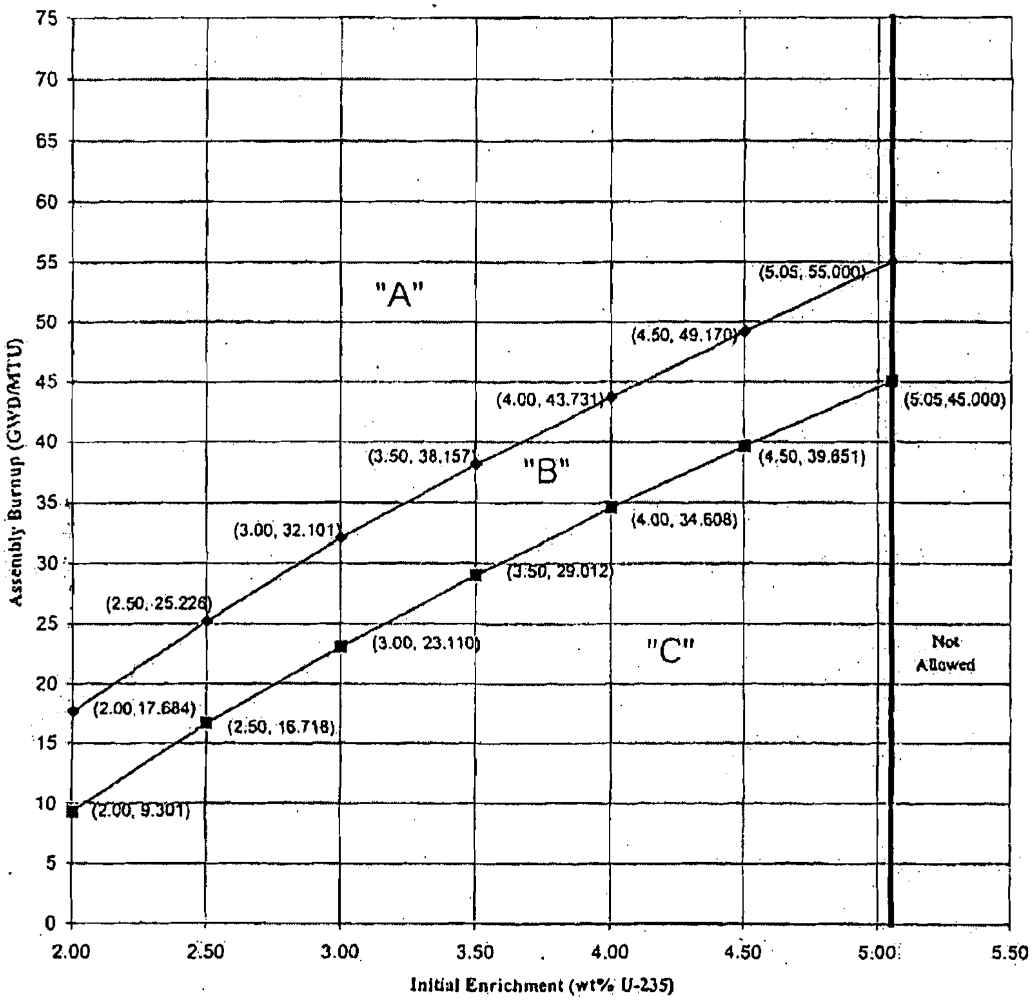

Figure 3.7.16-1 (page 1 of 1)
Burnup versus Enrichment Curve for
Spent Fuel Pool Storage Racks

NOTE: Fuel assemblies with initial enrichments less than 2.0 wt% U-235 will conservatively be required to meet the burnup requirements of 2.0 wt% U-235 assemblies. The approved loading patterns applicable to Category "A," "B," and "C" assemblies are specified in the Bases.

3.7 PLANT SYSTEMS

3.7.17 Secondary Specific Activity

LCO 3.7.17 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.18 Steam Generator Level

LCO 3.7.18 Water Level of each steam generator shall be:

a. Less than or equal to the maximum water level shown in Figure 3.7.18-1 when in MODE 1 or 2;

b. ≤ 96% Operate Range with LCO 3.3.11, "Steam and Feedwater Rupture Control System (SFRCS) Instrumentation," Function 1 (Main Steam Line Pressure - Low) not bypassed when in MODE 3;

c. ≤ 96% Operate Range with LCO 3.3.11, Function 1 bypassed and both main feedwater (MFW) pumps not capable of supplying feedwater to the steam generators when in MODE 3; and

d. ≤ 50 inches Startup Range with LCO 3.3.11, Function 1 bypassed and one or both MFW pumps capable of supplying feedwater to the steam generators when in MODE 3.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Enter applicable Conditions and Required Actions of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," when high steam generator water level results in exceeding the SDM limits.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Water level in one or more steam generators not within limits.	A.1	Restore steam generator level to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.18.1	Verify steam generator water level to be within limits.	12 hours

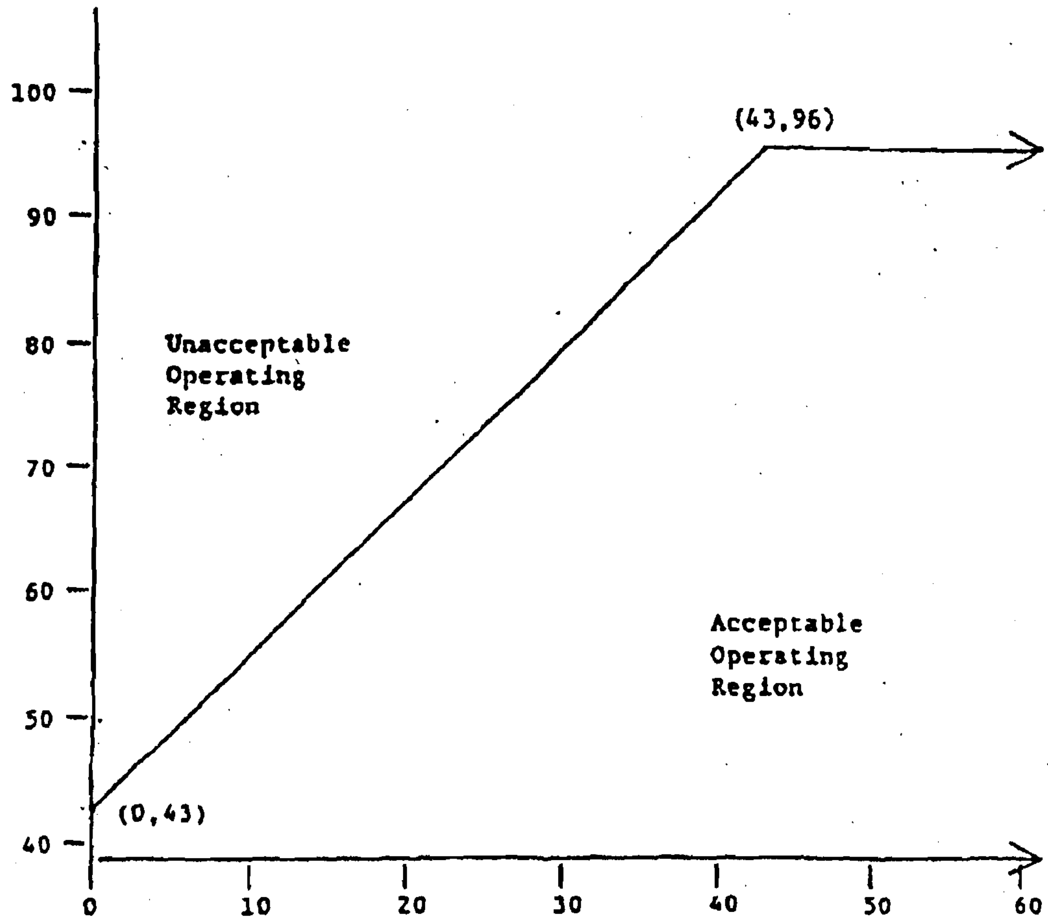

Figure 3.7.18-1 (page 1 of 1)
Maximum Allowable Steam Generator Level

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Two emergency diesel generators (EDGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System; and

c. Two load sequencers for Train 1 and two load sequencers for Train 2.

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

ACTIONS

-----NOTE-----

LCO 3.0.4.b is not applicable to EDGs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore offsite circuit to OPERABLE status.	72 hours
B. One EDG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable EDG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE EDG is not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE EDG.	24 hours
	AND		
	B.4	Restore EDG to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours
D. One offsite circuit inoperable. AND One EDG inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any train. ------------------------------	
	D.1 Restore offsite circuit to OPERABLE status.	12 hours
	OR	
	D.2 Restore EDG to OPERABLE status.	12 hours
E. Two EDGs inoperable.	E.1 Restore one EDG to OPERABLE status.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours
G. -----------NOTE----------- Separate Condition entry is allowed for each train. --------------------------- One or more trains with one load sequencer inoperable.	G.1 Remove inoperable load sequencer.	1 hour
H. -----------NOTE----------- Separate Condition entry is allowed for each train. --------------------------- Required Action and associated Completion Time of Condition G not met. OR One or more trains with two load sequencers inoperable.	H.1 Declare associated EDG inoperable.	Immediately
I. Three or more AC sources inoperable.	I.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	-----NOTES----- 1. All EDG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. A modified EDG start involving idling and/or gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.8 must be met. ---------- Verify each EDG starts from standby conditions and achieves steady state voltage ≥ 3744 V and ≤ 4400 V, and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.	31 days
SR 3.8.1.3	-----NOTES----- 1. EDG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one EDG at a time. 4. This SR shall be preceded by and immediately follow, without shutdown, a successful performance of SR 3.8.1.2 or SR 3.8.1.8. ---------- Verify each EDG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2340 kW and ≤ 2600 kW.	31 days
SR 3.8.1.4	Verify each day tank contains ≥ 4000 gal of fuel oil.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify interval between each sequenced load block is within ± 10% of design interval for each emergency load sequencer and each emergency time delay relay.	31 days
SR 3.8.1.7	Verify the fuel oil transfer system operates to transfer fuel oil from fuel oil storage tank to the day tank.	92 days
SR 3.8.1.8	-----NOTE----- All EDG starts may be preceded by an engine prelube period. ---------- Verify each EDG starts from standby condition and achieves: a. In ≤ 10 seconds, voltage ≥ 4031 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3744 V and ≤ 4400 V, and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.	184 days

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.9	-------------------NOTES------------------- 1. SR 3.8.1.9.a is only required to be met when the unit auxiliary source is supplying the electrical power distribution subsystem. 2. The automatic transfer portion of SR 3.8.1.9.a and all of SR 3.8.1.9.b shall not normally be performed in MODE 1 or 2. However, they may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --	
	Verify automatic and manual transfer of AC power sources from: a. The unit auxiliary source to the pre-selected offsite circuit; and b. The normal offsite circuit to the alternate offsite circuit.	24 months

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.10	-------------------NOTES------------------- 1. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with the EDG synchronized with offsite power, it shall be performed within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each EDG rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is ≤ 66.75 Hz.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	-------NOTES------- 1. All EDG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --------------- Verify on an actual or simulated loss of offsite power signal: a. De-energization of essential buses; b. Load shedding from essential buses; and c. EDG auto-starts from standby condition and: 1. Energizes permanently connected loads in ≤ 10 seconds; 2. Energizes auto-connected shutdown loads through individual time delay relays; 3. Maintains steady-state voltage ≥ 3744 V and ≤ 4400 V; 4. Maintains steady-state frequency ≥ 59.5 Hz and ≤ 60.5 Hz; and 5. Supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-----NOTE----- This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify each EDG's noncritical automatic trips are bypassed on actual or simulated loss of voltage signal on the essential bus or an actual or simulated Safety Features Actuation System (SFAS) actuation signal.	 24 months
SR 3.8.1.13	-----NOTES----- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If performed with EDG synchronized with offsite power, it shall be performed within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---------- Verify each EDG operates for ≥ 8 hours: a. For ≥ 2 hours loaded ≥ 2730 kW and ≤ 2860 kW; and b. For the remaining hours of the test loaded ≥ 2340 kW and ≤ 2600 kW.	 24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	-------------------NOTES------------------- 1. This Surveillance shall be performed within 5 minutes of shutting down the EDG after the EDG has operated ≥ 1 hour loaded ≥ 2340 kW and ≤ 2600 kW. Momentary transients outside of load range do not invalidate this test. 2. All EDG starts may be preceded by an engine prelube period. ---	
	Verify each EDG starts and achieves: a. In ≤ 10 seconds, voltage ≥ 4031 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3744 V and ≤ 4400 V, and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	----------NOTES---------- 1. All EDG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. -------------------------------	
	Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SFAS actuation signal: a. De-energization of essential buses; b. Load shedding from essential buses; c. EDG auto-starts from standby condition and: 1. Energizes permanently connected loads in $\leq$ 10 seconds; 2. Energizes auto-connected emergency loads through load sequencer and individual time delay relays; 3. Achieves steady-state voltage $\geq$ 3744 V and $\leq$ 4400 V; 4. Achieves steady-state frequency $\geq$ 59.5 Hz and $\leq$ 60.5 Hz; and 5. Supplies permanently connected and auto-connected emergency loads for $\geq$ 5 minutes.	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown"; and

b. One emergency diesel generator (EDG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	---------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. ------------------------------	
	A.1 Declare affected required feature(s) with no offsite power available inoperable. <u>OR</u>	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.3 Initiate action to restore required offsite circuit to OPERABLE status.	Immediately
B. One required EDG inoperable.	B.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.3 Initiate action to restore required EDG to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	-----NOTE----- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.10, SR 3.8.1.11, SR 3.8.1.12, SR 3.8.1.13, and SR 3.8.1.14. ----------	
	For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources - Operating," except SR 3.8.1.6, SR 3.8.1.9, and SR 3.8.1.15, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required emergency diesel generator (EDG).

APPLICABILITY: When associated EDG is required to be OPERABLE.

ACTIONS

-NOTE-

Separate Condition entry is allowed for each EDG.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more EDGs with fuel level < 32,000 gal and > 26,800 gal in storage tank.	A.1	Restore fuel oil level to within limits.	48 hours
B. One or more EDGs with lube oil inventory < 260 gal and > 236 gal.	B.1	Restore lube oil inventory to within limits.	48 hours
C. One or more EDGs with stored fuel oil total particulates not within limit.	C.1	Restore fuel oil total particulates to within limits.	7 days
D. One or more EDGs with new fuel oil properties not within limits.	D.1	Restore stored fuel oil properties to within limits.	30 days
E. One or more EDGs with required starting air receiver pressure < 210 psig and ≥ 139 psig.	E.1	Restore starting air receiver pressure to ≥ 210 psig.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR One or more EDGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated EDG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains $\geq$ 32,000 gal of fuel.	31 days
SR 3.8.3.2	Verify lube oil inventory for each EDG is $\geq$ 260 gal.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each required EDG air start receiver pressure is $\geq$ 210 psig.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Train 1 and Train 2 DC electrical power sources shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two required battery chargers on one train inoperable.	A.1	Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND		
	A.2	Verify battery float current $\leq$ 2 amps.	Once per 12 hours
	AND		
	A.3	Restore required battery charger(s) to OPERABLE status.	72 hours
B. One DC electrical power source inoperable for reasons other than Condition A.	B.1	Restore DC electrical power source to OPERABLE status.	2 hours
C. Required Action and Associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days
SR 3.8.4.2	Verify each required battery charger supplies ≥ 475 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. <u>OR</u> Verify each required battery charger can recharge the battery to the fully charged state within 12 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	18 months
SR 3.8.4.3	-------------------NOTES------------------- 1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of SR 3.8.4.3. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. Credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required actual or simulated emergency loads for the design duty cycle when subjected to a battery service test.	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 One Train 1 or Train 2 DC electrical power source shall be OPERABLE to support one train of the DC Electrical Power Distribution System required by LCO 3.8.10, "Distribution System - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

NOTE

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required DC electrical power source inoperable.	A.1	Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>		
	A.2	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>		
	A.3	Initiate action to restore required DC electrical power source to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-----NOTE----- The following SR is not required to be performed: SR 3.8.4.3. ---------- For the DC source required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.2 SR 3.8.4.3	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Parameters

LCO 3.8.6 Battery parameters for the Train 1 and Train 2 batteries shall be within limits.

APPLICABILITY: When associated DC electrical power sources are required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cells float voltage ≤ 2.07 V.	A.1	Perform SR 3.8.4.1	2 hours
	<u>AND</u>		
	A.2	Perform SR 3.8.6.1.	2 hours
	<u>AND</u>		
	A.3	Restore affected cell voltage > 2.07 V.	24 hours
B. One or more batteries with float current > 2 amps.	B.1	Perform SR 3.8.4.1.	2 hours
	<u>AND</u>		
	B.2	Restore battery float current to ≤ 2 amps.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ----------NOTE---------- Required Action C.2 shall be completed if electrolyte level was below the top of plates. ----------------------------	---------------NOTE--------------- Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of plates. ------------------------------------		
One or more batteries with one or more cells electrolyte level less than minimum established design limits.	C.1	Restore electrolyte level to above top of plates.	8 hours
	<u>AND</u>		
	C.2	Verify no evidence of leakage.	12 hours
	<u>AND</u>		
	C.3	Restore electrolyte level to greater than or equal to minimum established design limits.	31 days
D. One or more batteries with pilot cell electrolyte temperature less than minimum established design limits.	D.1	Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours
E. Batteries in redundant trains with battery parameters not within limits.	E.1	Restore battery parameters for batteries in one train to within limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR One or more batteries with one or more battery cells float voltage ≤ 2.07 V and float current > 2 amps. OR SR 3.8.6.6 not met.	F.1	Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	-----NOTE----- Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. ----- Verify each battery float current is ≤ 2 amps.	7 days
SR 3.8.6.2	Verify each battery pilot cell voltage is > 2.07 V.	31 days
SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days
SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	31 days

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.6.5	Verify each battery connected cell voltage is > 2.07 V.	92 days
SR 3.8.6.6	------NOTE------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. Credit may be taken for unplanned events that satisfy this SR. ------------ Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months AND 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 The Train 1 and Train 2 inverters shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One inverter inoperable.	A.1	-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any 120 VAC vital bus de-energized. ---------- Restore inverter to OPERABLE status.	24 hours
B. Two inverters in one train inoperable.	B.1	Restore one inverter to OPERABLE status	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify, for each inverter, correct inverter voltage, frequency, and alignment to the associated 120 VAC vital bus.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 One inverter shall be OPERABLE to support the 120 VAC vital electrical distribution subsystem required by LCO 3.8.10, "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore required inverter to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify, for the required inverter, correct inverter voltage, frequency, and alignment to the associated 120 VAC vital bus.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Train 1 and Train 2 AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	------NOTE------ Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for DC sources made inoperable by inoperable power distribution subsystems. ------------		
	A.1	Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours
B. One or more AC vital buses inoperable.	B.1	Restore AC vital bus(es) to OPERABLE status.	8 hours
C. One DC electrical power distribution subsystems inoperable.	C.1	Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portions of AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.3 Initiate actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Declare associated required decay heat removal subsystem(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System (RCS) and the refueling canal shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

-----NOTE-----
Only applicable to the refueling canal when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions.	Immediately
	AND	
	A.2 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1	Suspend positive reactivity additions, except the introduction of coolant into the RCS.	Immediately
	AND		
	A.2	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1	Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND		
	B.2	Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2	-------------------------NOTE------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.3 Decay Time

LCO 3.9.3 The reactor shall be subcritical for ≥ 72 hours.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor not subcritical for ≥ 72 hours.	A.1 Suspend movement of irradiated fuel assemblies within the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify reactor subcritical for ≥ 72 hours.	Prior to movement of irradiated fuel assemblies within the reactor pressure vessel

3.9 REFUELING OPERATIONS

3.9.4 Decay Heat Removal (DHR) and Coolant Circulation - High Water Level

LCO 3.9.4 One DHR loop shall be OPERABLE and in operation.

-------------------------------NOTE-------------------------------
The required DHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, "Boron Concentration."

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DHR loop requirements not met.	A.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy DHR loop requirements.	Immediately
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close equipment hatch and secure with four bolts.	4 hours
	AND		
	A.5	Close one door in each air lock.	4 hours
	AND		
	A.6	Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by a Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one DHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 2800 gpm.	12 hours

3.9 REFUELING OPERATIONS

3.9.5 Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level

LCO 3.9.5 Two DHR loops shall be OPERABLE, and one DHR loop shall be in operation.

-----NOTES-----

1. All DHR pumps may be removed from operation for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10 degrees F below saturation temperature;

   b. No operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, "Boron Concentration;" and

   c. No draining operations to further reduce RCS water volume are permitted.

2. One required DHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other DHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than required number of DHR loops OPERABLE.	A.1 Initiate action to restore DHR loop to OPERABLE status.	Immediately
	<u>OR</u>	
	A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No DHR loop OPERABLE or in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	B.2 Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately
	AND	
	B.3 Close equipment hatch and secure with four bolts.	4 hours
	AND	
	B.4 Close one door in each air lock.	4 hours
	AND	
	B.5 Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by a Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one DHR loop is in operation.	12 hours
SR 3.9.5.2	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to the required DHR pump that is not in operation.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 23 ft above the top of the reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling canal water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling canal water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

4.0 DESIGN FEATURES

4.1 Site Location

The Davis-Besse Nuclear Power Station is located on Lake Erie in Ottawa County, Ohio, approximately six miles northeast from Oak Harbor, Ohio and 21 miles east from Toledo, Ohio. The exclusion area boundary has a minimum radius of 2400 feet from the center of the plant.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy M5 or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy 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 fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rods

The reactor core shall contain 53 CONTROL RODS and 8 APSRs. The material shall be silver indium cadmium for the CONTROL RODS and inconel for the APSRs, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel pool storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

b. A nominal 9.22 inch center to center distance between fuel assemblies; and

c. Fuel assemblies stored in the spent fuel storage racks in accordance with LCO 3.7.16.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ when immersed in a hydrogenous mist, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 <u>Drainage</u>

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 9 feet above the top of the spent fuel storage racks.

4.3.3 <u>Capacity</u>

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1624 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affects nuclear safety.

5.1.2 The shift manager shall be responsible for the control room command function. During any absence of the shift manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Operator license shall be designated to assume the control room command function. During any absence of the shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active Senior Operator license or Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

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 safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the UFSAR.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer 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.

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned if the reactor contains fuel and an additional non-licensed operator shall be assigned if the reactor is operating in MODES 1, 2, 3, or 4;

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order 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;

5.2 Organization

5.2.2 Unit Staff (continued)

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position;

d. Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

   The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviation from the above guidelines shall be authorized in advance by the plant manager or designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

   Controls shall be included in the procedures such that the individual overtime shall be reviewed monthly by the plant manager or designee to ensure that excessive hours have not been assigned;

e. The operations manager shall either hold or have held a Senior Operator license. The assistant operations manager shall hold a Senior Operator license for the Davis-Besse Nuclear Power Station; and

f. When the reactor is operating in MODE 1, 2, 3, or 4 an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the radiation protection manager and the operations manager. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The operations manager shall be qualified as required by Specification 5.2.2.e.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The 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 and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.1 and Specification 5.6.2.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s); and

   b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. 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 (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include makeup, letdown, seal injection, seal return, low pressure injection, containment spray, high pressure injection, waste gas, primary sampling, and reactor coolant drain 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 least once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Radioactive Effluent Controls Program

This program conforms to 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 shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20;

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;

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 10 CFR 50, Appendix I;

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. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

5.5 Programs and Manuals

5.5.3 Radioactive Effluent Controls Program (continued)

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas beyond the site boundary shall be in accordance with Appendix B, Table 2, Column 1 to 10 CFR 20;

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 10 CFR 50, Appendix 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 > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies.

5.5.4 Reactor Vessel Internals Vent Valves Program

A program shall be established to implement the testing of the reactor vessel internals vent valves every 24 months as follows:

a. Verify by visual inspection that the valve body and valve disc exhibit no abnormal degradation;

b. Verify the valve is not stuck in an open position; and

c. Verify by manual actuation that the valve is fully open when a force of ≤ 400 lbs is applied vertically upward.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Vessel Internals Vent Valves Program test Frequencies.

5.5 Programs and Manuals

5.5.5 Allowable Operating Transient Cycles Program

This program provides controls to track the UFSAR, Section 5, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Inservice inspection of each reactor coolant pump flywheel shall be performed every 10 years. The inservice inspection shall be either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or a surface examination of exposed surfaces of the disassembled flywheel. The recommendations delineated in Regulatory Positions C.4.b(3), (4), and (5) of Regulatory Guide 1.14, Revision 1, August 1975, shall apply.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Codes) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

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

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG except during a steam generator tube rupture.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria:

1. Tubes found by inservice inspection to contain flaws, in a region of the tube that contains no repair, with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired;

2. Sleeves found by inservice inspection to contain flaws, in a region of the sleeve that contains no sleeve joint, with a depth equal to or exceeding 40% of the nominal sleeve wall thickness shall be plugged;

3. Tubes with a flaw, in either parent tube or the sleeve, within a sleeve to tube joint shall be plugged; and

4. Tubes with a flaw in a repair roll shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. For tubes that have undergone repair rolling, the tube and tube roll, outboard of the new roll area in the tube sheet, can be excluded from inspections because it is no longer part of the pressure boundary once the repair roll is installed. For tubes that have undergone sleeving repairs, the segment of the parent tube between the upper-most sleeve roll and the top of the middle sleeve roll can be excluded from inspection because it is no longer part of the pressure boundary once the sleeve is installed. In addition to meeting the requirements of d.1 through d.5 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

4. During each periodic SG tube inspection, inspect 100% of the tubes that have been repaired by the repair roll process. This special inspection shall be limited to the repair roll joint and the roll transitions of the roll repair.

5. Inspect peripheral tubes in the vicinity of the secured internal auxiliary feedwater header between the upper tube sheet and the 15th tube support plate during each periodic SG tube inspection. The tubes selected for inspection shall represent the entire circumference of the steam generator and shall total at least 150 peripheral tubes.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

1. Sleeving in accordance with Topical Report BAW-2120P.

2. Repair rolling in accordance with Topical Report BAW-2303P, Revision 4. The new roll area must be free of flaws in order for the repair to be considered acceptable.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

g. Special visual inspections: Visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and the external header thermal sleeves shall be performed on each SG through the auxiliary feedwater injection penetrations. These inspections shall be performed during the third period of each 10 year Inservice Inspection Interval (ISI).

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of safety related filter ventilation systems in accordance with Regulatory Guide 1.52, Revision 2, ANSI/ASME N510-1980, and ASTM D 3803-1989.

a. Demonstrate for each of the safety related systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below.

Safety Related Ventilation System	Flowrate (cfm)
Station Emergency Ventilation System (EVS)	≥ 7200 and ≤ 8800
Control Room Emergency Ventilation System (CREVS)	≥ 2970 and ≤ 3630

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (continued)

b. Demonstrate for each of the safety related systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below.

Safety Related Ventilation System	Flowrate (cfm)
Station EVS	≥ 7200 and ≤ 8800
CREVS	≥ 2970 and ≤ 3630

c. Demonstrate for each of the safety related 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 (86°F) and the relative humidity (RH) specified below.

Safety Related Ventilation System	Penetration (%)	RH (%)
Station EVS	≤ 2.5	95
CREVS	≤ 2.5	70

d. Demonstrate for each of the safety related systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below.

Safety Related Ventilation System	Delta P (inches wg)	Flowrate (cfm)
Station EVS	< 6	≥ 7200 and ≤ 8800
CREVS	< 4.4	≥ 2970 and ≤ 3630

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

5.5 Programs and Manuals

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid storage tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tank's contents.

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.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

    1. An API gravity or an absolute specific gravity within limits;

    2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil; and

    3. A clear and bright appearance with proper color, or a water and sediment content within limits;

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program (continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing Frequencies.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (continued)

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable; and

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.14.b.1 and 5.5.14.b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the 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-Test Program," dated September, 1995, as modified by the following exceptions:

1. A reduced duration Type A test may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

2. The fuel transfer tube blind flanges (containment penetrations 23 and 24) will not be eligible for extended test frequencies. Their Type B test frequency will remain at 30 months. However, as-found testing will not be required.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 38 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $< 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.015\ L_a$ when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq 0.01\ L_a$ when the volume between the door seals is pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.16 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, including the following:

a. Actions to restore battery cells with float voltage < 2.13 V;

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; and

c. Actions to verify that the remaining cells are > 2.07 V when a pilot cell or cells have been found to be < 2.13 V.

5.5.17 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program (continued)

occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary;

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance;

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Section C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0;

d. Measurements, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary;

e. The quantitative limits on unfiltered air leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in Specification 5.5.17.c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis; and

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by Specifications 5.5.17.c and 5.5.17.d, respectively.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 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 by May 15 of each year. The report shall include summaries, interpretations, and analyses 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 the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted in accordance with 10 CFR 50.36a. 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 consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. SL 2.1.1.1, "Reactor Core Safety Limits";

2. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";

3. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";

4. LCO 3.1.7, "Position Indicator Channels," (SR 3.1.7.1 limits);

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. LCO 3.1.8, "PHYSICS TEST Exceptions - MODE 1";

6. LCO 3.1.9, "PHYSICS TEST Exceptions - MODE 2";

7. LCO 3.2.1, "Regulating Rod Insertion Limits";

8. LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";

9. LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits";

10. LCO 3.2.4, "QUADRANT POWER TILT (QPT)";

11. LCO 3.2.5, "Power Peaking Factors";

12. LCO 3.3.1,"Reactor Protection System (RPS) Instrumentation," Function 8 (Flux - ΔFlux – Flow) Allowable Value; and

13. LCO 3.9.1, "Boron Concentration."

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," or any other new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A. The applicable approved revision number for BAW-10179P-A at the time of the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT (COLR). The COLR shall also list any new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A.

c. As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of RTP is specified in a previously approved method, an actual value of 100.37% of RTP may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:

1. Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.

d. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

e. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," June 1986.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6 Reporting Requirements

5.6.6 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. Active degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism;

f. Total number and percentage of tubes plugged or repaired to date;

g. The results of condition monitoring, including the results of tube pulls and in-situ testing;

h. The effective plugging percentage for all plugging and tube repairs in each SG; and

i. Repair method utilized and the number of tubes repaired by each repair method.

5.6.7 Remote Shutdown System Report

When a report is required by Condition C of LCO 3.3.18, "Remote Shutdown System," a report shall be submitted within the following 30 days. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the control circuit or transfer switch of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess one of the following:

   1. A radiation monitoring device that continuously displays radiation dose rates in the area;

   2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint;

   3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

   4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door, gate, or other barrier that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee; and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess one of the following:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint;

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area; or

(ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where Specifications 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 279 TO FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated August 3, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072200451) as supplemented by the letters discussed below, FirstEnergy Nuclear Operating Company, et al. (the licensee) requested changes to the technical specifications (TSs) for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The proposed changes would revise the current TSs (CTS) to the improved TSs (ITS).

The seven supplemental letters to the application provided the following information for the proposed ITS Conversion:

- Letter from Barry S. Allen, Vice President, DBNPS, to the NRC Document Control Desk (DCD), dated May 16, 2008 (ADAMS Accession No. ML081480464), which supplements the licensee's application and provides revisions to the TS for Sections 3.0, 3.1, 3.2, 3.4, 3.6, and 4.0. The revisions to the TS for individual Sections 3.0, 3.1, 3.2, 3.4, 3.6, and 4.0 can be found in ADAMS as follows: Section 3.0 (ADAMS Accession No. ML081480465), Section 3.1 (ADAMS Accession No. ML081480466), Section 3.2 (ADAMS Accession No. ML081480467), Section 3.4 (ADAMS Accession No. ML081480468), Section 3.6 (ADAMS Accession No. ML081480469), and Section 4.0 (ADAMS Accession No. ML081480471).

- Letter from Barry S. Allen, Vice President, DBNPS, to the NRC DCD, dated May 16, 2008 (ADAMS Accession No. ML081430105), which provides a copy of the licensee's responses to NRC questions, on TS proposals for Sections 3.0, 3.1, 3.2, 3.4, 3.6, and 4.0, that took place via the NRC-DBNPS ITS Conversion web page discussed below.

- Letter from Barry S. Allen, Vice President, DBNPS, to the NRC DCD, dated July 23, 2008 (ADAMS Accession No. ML082070079), which provides responses to a request for additional information (RAI) letter from the NRC, dated June 20, 2008 (ADAMS Accession No. ML081650364).

- Letter from Barry S. Allen, Vice President, DBNPS, to the NRC DCD, dated August 7, 2008 (ADAMS Accession No. ML082270658), which supplements the licensee's application and provides revisions to the TS for Sections 1.0, 2.0, 3.3, 3.5, 3.7, 3.8, 3.9, 5.0, as well as revisions to the no significant hazards consideration (NSHC) and the Split Report Summary. The revisions to the TS for individual Sections 1.0, 2.0, 3.3, 3.5, 3.7, 3.8, 3.9, 5.0, as well as revisions to the NSHC and the Split Report Summary, can be found in ADAMS as follows: Section 1.0 (ADAMS Accession No. ML082270663), Section 2.0 (ADAMS Accession No. ML082270664), Section 3.3 (ADAMS Accession No. ML082270665), Section 3.5 (ADAMS Accession No. ML082270666), Section 3.7 (ADAMS Accession No. ML082270669), Section 3.8 (ADAMS Accession No. ML082270670), Section 3.9 (ADAMS Accession No. ML082270671), Section 5.0 (ADAMS Accession No. ML082270667), NSHC (ADAMS Accession No. ML082270662), and Split Report Summary (ADAMS Accession No. ML082270661).

- Letter from Barry S. Allen, Vice President, DBNPS, to the NRC DCD, dated September 3, 2008 (ADAMS Accession No. ML082490154), which provides responses to a RAI letter from the NRC, dated June 18, 2008 (ADAMS Accession No. ML081570588).

- Letter from Barry S. Allen, Vice President, DBNPS, to the NRC DCD, dated August 26, 2008 (ADAMS Accession No. ML082600594), which provides a copy of the licensee's responses to NRC questions, on TS proposals for Sections 1.0, 2.0, 3.3, 3.5, 3.7, 3.8, 3.9, 5.0, that took place via the NRC-DBNPS ITS Conversion web page discussed below.

- Letter from Barry S. Allen, Vice President, DBNPS, to the NRC DCD, dated October 21, 2008 (ADAMS Accession No. ML083010076), which provide the retyped copy of TS pages to be issued in this amendment.

The following safety evaluation (SE) on the proposed ITS Conversion is based on the application dated August 3, 2007, and the information provided to the NRC through the DBNPS ITS Conversion web page hosted by EXCEL Services Corporation and supplements provided, as discussed above. To expedite its review of the application, the NRC staff issued its RAIs through the DBNPS ITS Conversion web page and the licensee addressed the RAIs by providing responses on the web page. Entry into the database was protected so that only licensee and NRC reviewers could enter information into the database to add RAIs (NRC) or providing responses to the RAIs (licensee); however, the public could enter the database to only read the questions asked and the responses provided. To be in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.4 for written communications for license amendment requests (LARs) and to have the database on the DBNPS docket before the amendment was issued in accordance with 10 CFR 50.30 for oath and affirmation, the licensee submitted a copy of the database in a submittal to the NRC after there were no further RAIs.

The additional information provided in the seven supplemental letters, did not expand the scope of the application as noticed, and did not change the NRC staff's original notice published in the *Federal Register* on May 22, 2008 (73 FR 29787 - 29791).

2.0 BACKGROUND

DBNPS has been operating with the TSs issued with the original Facility Operating License dated April 22, 1977, as amended. The proposed conversion to the ITS is based upon:

- NUREG-1430, "Standard Technical Specifications (STS) Babcock and Wilcox Plants," Rev. 3.0;
- DBNPS CTS;
- "Final Policy Statement [FPS] on Technical Specification Improvements for Nuclear Power Reactors," published on July 22, 1993 (58 FR 39132); and
- Part 50 of 10 CFR, Section 50.36, "Technical Specifications."

Hereinafter, the proposed TSs for DBNPS are referred to as the ITS, the existing TSs are referred to as the CTS, and the improved standard TSs (ISTS), given in NUREG-1430, are referred to as the ISTS. The corresponding Bases are ITS Bases, CTS Bases, and ISTS Bases, respectively. For convenience, a list of acronyms used in this SE is provided in Attachment 1, to this SE.

In addition to basing the ITS on the ISTS, the FPS, and the requirements in 10 CFR 50.36, the licensee retained portions of the CTS as a basis for the ITS. During the course of its review, the NRC staff utilized DBNPS ITS Conversion database, issued several RAIs, and conducted a series of telephone conference calls with the licensee. The Conversion database, RAIs, and conference calls served to clarify the ITS with respect to the guidance in the FPS and the ISTS. The NRC staff requested that the licensee docket the DBNPS ITS conversion database in a sworn statement with regards to its accuracy, as well as docket all RAIs and responses under oath and affirmation, in a supplement to the license amendment. The licensee also proposed changes of a generic nature that were not in the ISTS. The NRC staff requested that the licensee submit such generic changes as proposed changes to the ISTS through the industry TSs Task Force (TSTF). These generic issues were considered for specific applications in the DBNPS ITS. Consistent with the Commission's FPS and 10 CFR 50.36, the licensee proposed transferring some CTS requirements to licensee-controlled documents (such as the DBNPS Updated Final Safety Analysis Report (UFSAR)), for which changes to the documents by the licensee are controlled by a regulation (e.g. 10 CFR 50.59) and which may be made without prior NRC approval. NRC-controlled documents, such as the TSs, may not be changed by the licensee without prior NRC approval. In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS, and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The overall objective of the proposed amendment, consistent with the FPS, is to rewrite, reformat, and streamline the DBNPS CTS to provide clearer, more readily understandable requirements to ensure safer operation of the units, while still satisfying the requirements of 10 CFR 50.36. During its review, the NRC staff relied on the FPS and 10 CFR 50.36, and the ISTS as guidance for acceptance of CTS changes. This SE provides a summary basis for the NRC staff's conclusion that use of the licensee's proposed ITS based on ISTS, as modified by plant-specific changes, is acceptable for continued operation of DBNPS. This SE also explains the NRC staff's conclusion that the ITS are consistent with the DBNPS CLB and the requirements of 10 CFR 50.36.

The license conditions included in the proposed amendment will make enforceable the following aspects of the conversion: (1) the schedule for the first performance of new and revised surveillance requirements (SRs); and (2) the relocation of CTS requirements into licensee-controlled documents as part of the implementation of the ITS.

For the reasons stated *infra* in this SE, the NRC staff finds that the ITS issued with this LAR complies with Section 182a of the Atomic Energy Act, 10 CFR 50.36, and the guidance in the FPS, and that they are in accordance with the common defense and security and provide adequate protection of the health and safety of the public.

3.0 REGULATORY REQUIREMENTS

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses state:

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences. As recorded in the Statements of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports" (33 FR 18610; December 17, 1968), the Commission noted that applicants were expected to incorporate into their TSs "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits (SLs), limiting safety system settings (LSSSs), and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TSs.

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TSs. On February 6, 1987, the Commission issued an interim policy statement (IPS) on TS improvements, "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, utility owners groups and the NRC staff developed ISTSs (e.g., NUREG-1430) that would establish model TSs based on the Commission's policy for each primary reactor type. In addition, the NRC staff, licensees, and owners groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing

TSs, which gives appropriate consideration to human factors engineering principles and was used throughout the development of plant-specific ITS.

In September 1992, the Commission issued NUREG-1430, Rev. 0, which was developed using the guidance and criteria contained in the Commission's IPS. The ISTSs in NUREG-1430 were established as a model for developing the ITSs for Babcock and Wilcox-type plants, in general. The ISTSs reflect the results of a detailed review of the application of the IPS criteria which have been incorporated in 10 CFR 50.36(c)(2)(ii), to generic system functions, which were published in a "Split Report" issued to the nuclear steam supply system vendor owners groups in May 1988. ISTSs also reflect the results of extensive discussions concerning various drafts of ISTSs so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all reactor designs. As such, the generic Bases presented in NUREG-1430 provide an abundance of information regarding the extent to which the ISTSs present requirements that are necessary to protect public health and safety. The ISTSs in NUREG-1430, Rev. 3.0, as modified, apply to DBNPS.

On July 22, 1993, the Commission issued its FPS, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Atomic Energy Act and 10 CFR 50.36. The FPS described the safety benefits of the ISTSs and encouraged licensees to use the ISTSs as the basis for plant-specific TS amendments and for complete conversions to ITSs based on the ISTSs. In addition, the FPS gave guidance for evaluating the required scope of the TSs and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TSs. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TSs, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the [Atomic Energy] Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the FPS should be retained in the TSs; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are stated as follows:

Criterion 1 — Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident [DBA] or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3 A structure, system, or component [SSC] that is part of the primary success path and which functions or actuates to mitigate a [DBA] or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4 A structure, system, and component [SSC] which operating experience or probabilistic risk assessment [PRA] has shown to be significant to public health and safety.

Part 4.0 of this SE explains the NRC staff's determination that the conversion of the DBNPS CTSs to ITSs based on ISTSs, as modified by plant-specific changes, is consistent with the DBNPS, CLB, the requirements and guidance of the FPS, and 10 CFR 50.36.

4.0 EVALUATION

In its review of the DBNPS ITS application, the NRC staff evaluated five kinds of CTS changes as defined by the licensee. The NRC staff's review also included an evaluation of whether existing regulatory requirements are adequate for controlling future changes to requirements that are removed from the CTSs and placed in licensee-controlled documents. The following are the five types of CTS changes, as stated by the licensee:

A Administrative - Changes to the CTSs that do not result in new requirements or change operational restrictions and flexibility.

M More Restrictive - Changes to the CTSs that result in added restrictions or reduced flexibility.

L Less Restrictive - Changes to the CTSs that result in reduced restrictions or added flexibility.

LA Removed Details - Changes to the CTSs that eliminate detail and relocate the detail to a licensee-controlled document. Typically, this involves details of system design and function, or procedural detail on methods of conducting a [SR]. These changes are supported in aggregate by a single generic NSHC. In addition, the generic type of removed detail change is identified in italics at the beginning of the discussion of change [DOC].

R Relocated Specifications - Changes to the CTSs that relocate the requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii).

The ITS application included a justification for each proposed change to the CTSs in a numbered DOC, using the above letter designations as appropriate. In addition, the ITS application included an explanation of each difference between ITS and ISTS requirements in a numbered justification for deviation.

The changes to the CTSs, as presented in the ITS application, are listed and described in the following five tables (for each ITS section) provided as Attachments 2 through 6 to this SE:

- Table A - Administrative Changes
- Table M - More Restrictive Changes
- Table L - Less Restrictive Changes
- Table LA - Removed Detail Changes
- Table R - Relocated Specifications

These tables provide a summary description of the proposed changes to the CTSs. The tables are only meant to summarize the changes being made to the CTSs. The details as to what the actual changes are and how they are being made to the CTSs or ITSs are provided in the licensee's application and supplemental letters.

The NRC staff's evaluation and additional description of the kinds of changes to the CTS requirements listed in Tables A, M, L, LA, and R attached to this SE are presented in Sections A through E below, as follows:

- Section A - Administrative Changes
- Section B - More Restrictive Changes
- Section C - Less Restrictive Changes
- Section D - Removed Details
- Section E - Relocated Specifications

The control of specifications, requirements, and information relocated from the CTSs to licensee-controlled documents is described in Section F below, and other CTS changes (i.e., beyond-scope changes, changes beyond the scope of a TS Conversion) are described in Section G below.

A. <u>Administrative Changes to the CTS</u>

Administrative changes are intended to incorporate human factors principles into the form and structure of the ITSs so that plant operations personnel can use them more easily. These changes are editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions. Every section of the ITSs reflects this type of change. In order to ensure consistency, the NRC staff review of the licensee proposed TS used the ISTSs as guidance to reformat and make other administrative changes that do not involve technical changes to CTSs. Among the changes proposed by the licensee and found acceptable by the NRC staff are:

- Identifying plant-specific wording for system names, etc.;
- Splitting up requirements currently grouped under a single CTS and moving them to more appropriate locations in two or more specifications of the ITSs;

- Combining related requirements currently presented in separate specifications of the CTSs into a single specification of ITSs;
- Presentation changes that involve rewording or reformatting for clarity (including moving an existing requirement to another location within the TSs) but that do not involve a change in requirements;
- Wording changes and additions that are consistent with CTS interpretation and practice and that more clearly or explicitly state existing requirements;
- Deletion of TSs that no longer apply;
- Deletion of details that are strictly informational and have no regulatory basis; and,
- Deletion of redundant TS requirements that exist elsewhere in the TSs.

Table A attached to this SE lists the administrative changes being made in the DBNPS ITS conversion. Table A is organized in ITS order by each A-type DOC to the CTSs, provides a summary description of the administrative change that was made, and provides CTS and ITS references. The NRC staff reviewed all of the administrative and editorial changes proposed by the licensee and finds them acceptable because they are compatible with the Writer's Guide and the ISTSs, do not result in any change in operating requirements, and are consistent with the Commission's regulations.

B. More Restrictive Changes to the CTS

The licensee, in electing to implement the specifications of the ISTSs, proposed a number of requirements that are more restrictive than those in the CTSs. The ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTSs, or have additional restrictions that are not in the CTSs, but are in the ISTSs. Among the changes proposed by the licensee and found acceptable by the NRC staff are:

- Placement of an LCO on plant equipment that is not required by the CTSs;
- More restrictive requirements to restore inoperable equipment;
- More restrictive SRs.

Table M attached to this SE lists the more restrictive changes being made in the DBNPS ITS conversion. Table M is organized in ITS order by each M-type DOC to the CTSs, provides a summary description of each more restrictive change that was adopted, and references the affected CTSs and ITSs. These changes are additional restrictions on plant operation that enhance safety and are acceptable.

C. Less Restrictive Changes to the CTS

Less restrictive requirements include deletions of and relaxations to portions of the CTS requirements that are being retained in the ITSs. When requirements have been shown to give little or no safety benefit, their relaxation or removal from the TSs may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups' comments on ISTSs. The NRC staff reviewed generic relaxations contained in the ISTSs and found them acceptable because they are consistent with current licensing practices and the

Commission's regulations. The DBNPS design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in the ISTSs and thus provide a basis for ITSs.

All of the less restrictive changes to the CTSs have been evaluated and found to involve deletions of and relaxations to portions of CTS requirements that can be grouped in the following 10 categories:

Category 1 – Relaxation of LCO Requirement
Category 2 – Relaxation of Applicability
Category 3 – Relaxation of Completion Time
Category 4 – Relaxation of Required Action
Category 5 – Deletion of SR
Category 6 – Relaxation of SR Acceptance Criteria
Category 7 – Relaxation of Surveillance Frequency, Non-24 Month Type
Category 8 – Deletion of Reporting Requirements
Category 9 – Addition of LCO 3.0.4 Exception
Category 10 – Deletion of SR Shutdown Performance Requirements

The following discussion addresses why these categories of less restrictive changes are acceptable:

Category 1 – Relaxation of LCO Requirement

Certain CTS LCOs specify limits on operational and system parameters beyond those necessary to ensure meeting safety analysis assumptions and, therefore, are considered overly restrictive. The CTSs also contain operating limits that have been shown to give little or no safety benefit to the operation of the plant. The ITSs, consistent with the guidance in the ISTSs, would delete or revise such operating limits. CTS LCO changes of this type include: (1) redefining operating modes, including mode title changes; (2) deleting or revising operational limits to establish requirements consistent with applicable safety analyses; (3) deleting requirements for equipment or systems that establish system capability beyond that assumed to function by the applicable safety analyses, or that are implicit to the ITS requirement for systems, components, and devices to be operable; and (4) adding allowances to use administrative controls on plant devices and equipment during times when automatic control is required, or to establish temporary administrative limits, as appropriate, to allow time for systems to establish equilibrium operation. TSs changes represented by this type allow operators to more clearly focus on issues important to safety. The resultant ITS LCOs maintain an adequate degree of protection consistent with the safety analysis. They also improve focus on issues important to safety and provide reasonable operational flexibility without adversely affecting the safe operation of the plant. Changes involving the relaxation of LCOs are consistent with the guidance established by the ISTSs taking into consideration the DBNPS CLB. Therefore, based on the above, Category 1 changes are acceptable.

Category 2 – Relaxation of Applicability

The CTSs require compliance with the LCO during the applicable Mode(s) or other conditions specified in the Specification's Applicability statement. When CTS Applicability requirements are inconsistent with the applicable accident analyses assumptions for a system, subsystem, or component specified in the LCO, the licensee proposed to change the LCO to establish a consistent set of requirements in the ITSs. These modifications or deletions are acceptable because, during the operational or other conditions specified in the ITSs Applicability requirements, the LCOs are consistent with the applicable safety analyses. Changes involving relaxation of applicability requirements are consistent with the guidance established by the ISTSs, taking into consideration the DBNPS CLB. Therefore, based on the above, Category 2 changes are acceptable.

Category 3 – Relaxation of Completion Time

Upon discovery of a failure to meet an LCO, the TSs specify time limits for completing Required Actions of the associated TS Conditions. Required Actions establish remedial measures that must be taken within specified Completion Times. Completion Times specify limits on the duration of plant operation in a degraded condition. Incorporating longer Completion Times is acceptable because such Completion Times will continue to be based on the operability status of redundant TSs required features, the capacity and capability of remaining TS-required features, provision of a reasonable time for repairs or replacement of required features, vendor-developed standard repair times, and the low probability of a DBA occurring during the repair period. Changes involving relaxation of Completion Times are consistent with the guidance established by the ISTSs, taking into consideration the DBNPS CLB. Therefore, based on the above, Category 3 changes are acceptable.

Category 4 – Relaxation of Required Action

LCOs specify the lowest functional capability or performance level of equipment that is deemed adequate to ensure safe operation of the facility. When an LCO is not met, the CTSs specify actions to restore the equipment to its required capability or performance level, or to implement remedial measures providing an equivalent level of protection. Compared to CTS required actions, certain proposed ITS actions would result in extending the time period during which the licensee may continue to operate the plant with specified equipment inoperable. Upon expiration of this time period, further action, which may include shutting down the plant, is required. Changes of this type include providing an option to (1) isolate a system, (2) place equipment in the state assumed by the safety analysis, (3) satisfy alternate criteria, (4) take manual actions in place of automatic actions, (5) "restore to operable status" within a specified time frame, (6) place alternate equipment into service, or (7) use more conservative TS instrumentation actuation setpoints. The resulting ITS actions provide measures that adequately compensate for the inoperable equipment, and are commensurate with the safety importance of the inoperable equipment, plant design, and industry practice. Therefore, these action requirements will continue to ensure safe operation of the plant. Changes involving relaxations of action requirements are consistent with the guidance established by the ISTSs, taking into consideration the DBNPS CLB. Therefore, based on the above, Category 4 changes are acceptable.

Category 5 – Deletion of SR

The CTSs require maintaining LCO specified SSCs operable by meeting SRs in accordance with specified SR frequencies. This includes conducting tests to demonstrate that such SSCs are operable and that LCO specified parameters are within specified limits. When the test acceptance criteria and any specified conditions for the conduct of the test are met, the equipment is deemed operable. The changes of this category relate to deletion of CTS SRs, including deletion of an SR in its entirety, deletion of acceptance criteria, and deletion of the conditions required for performing the SR.

Deleting the SRs, including acceptance criteria and/or conditions for performing the SRs, for these items provides operational flexibility, consistent with the objective of the ISTSs, without reducing confidence that the equipment is operable. For example, the CTSs contain SRs that are not included in the ISTSs for a variety of reasons. This includes deletion of SRs for measuring values and parameters that are not necessary to meet ISTS LCO requirements. Also, the ISTSs may not include reference to specific acceptance criteria contained in the CTSs, because these acceptance criteria are not necessary to meet ISTS LCO requirements, or are defined in other licensee-controlled documents. The changes to SR acceptance criteria are acceptable because appropriate testing standards are retained for determining that the LCO required features are operable as defined by the ISTSs.

Deleting conditions for performing SRs includes not requiring testing of deenergized equipment (e.g., instrumentation channel checks) or equipment that is already performing its intended safety function (e.g., position verification of valves locked in their safety actuation position). This category also includes allowing verification of the position of valves in high radiation areas by administrative means. ITS Administrative Controls (ITS 5.7) regarding access to high radiation areas make the likelihood of mispositioning such valves small. Waiving performance of a surveillance under these conditions is acceptable because the equipment is already performing its intended safety function.

The deletion of these CTS SRs optimizes test requirements for the affected safety systems and increases operational flexibility. Changes involving relaxations of SRs, as described, are consistent with the guidance established by the ISTSs, taking into consideration the DBNPS CLB. Therefore, based on the above, Category 5 changes are acceptable.

Category 6 – Relaxation of SR Acceptance Criteria

Prior to placing the plant in a specified operational Mode or other condition stated in the applicability of a LCO, and in accordance with the specified SR time interval thereafter, the CTSs require establishing the operability of each LCO-required component by meeting the SRs associated with the LCO. This usually entails performance of testing to demonstrate the operability of the LCO-required components, or the verification that specified parameters are within LCO limits. A successful demonstration of operability requires meeting the specified acceptance criteria, as well as any specified conditions, for the conduct of the test. Relaxations of CTS SRs would include relaxing both the acceptance criteria and the conditions of performance. Also, the ITSs would permit the use of an actual, as well as a simulated, actuation signal to satisfy SRs for automatically actuated systems. This is acceptable because TS-required features cannot distinguish between an "actual" signal and a "test" signal. These relaxations of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility. These CTS SR relaxations are consistent with the guidance established by the ISTSs in consideration of the DBNPS CLB.

Category 7 – Relaxation of Surveillance Frequency, Non-24 Month Type

Prior to placing the plant in a specified operational Mode or other condition stated in the applicability of an LCO, and in accordance with the specified SR time interval (frequency) thereafter, the CTSs require establishing the operability of each LCO-required component by meeting the SRs associated with the LCO. This usually entails performance of testing to demonstrate the operability of the LCO-required components, or the verification that specified parameters are within LCO limits. A successful demonstration of operability requires meeting the specified acceptance criteria, as well as any specified conditions, for the conduct of the test, at a specified frequency based on the reliability and availability of the LCO-required components.

Relaxations of CTS SRs would include extending the interval between the SRs. This interval is the surveillance test interval (STI). These relaxations of CTS SR frequencies (or extending the STI) optimize test requirements for the affected safety systems and increase operational flexibility. These CTS SR frequency relaxations (or extending the STI) are consistent with the guidance established by the ISTSs in consideration of the DBNPS CLB.

Category 8 – Deletion of Reporting Requirements

The CTSs contain requirements that are redundant to reporting regulations in 10 CFR. For example, CTSs include requirements that a "Reportable Event" is any of those conditions specified in 10 CFR 50.73. However, consistent with the ISTSs, the ITSs would omit many of the CTS reporting requirements because the reporting requirements in the regulations cited do not need repeating in the TSs to ensure timely submission to the NRC. Therefore, Category 8 changes have no impact on the safe operation of the plant. Deletion of these requirements is beneficial because it reduces the administrative burden on the licensee and in turn allows increased attention to plant operations important to safety. Therefore, Category 8 changes have no impact on the safe operation of the plant and are acceptable.

Category 9 – Addition of LCO 3.0.4 Exception

The CTS precludes a change in MODES while relying on the Actions of a Specification. However, consistent with the ISTSs, the ITSs would allow entry into a Mode or other specified condition in the Applicability, even when an LCO is not met, provided: (a) the associated Actions to be entered permit continued operation in the Mode or other specified condition in the Applicability for an unlimited period of time; (b) the performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the Mode or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or (c) an allowance is stated in the individual value, parameter, or other Specification.

The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide (RG) 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." The results of the risk assessment shall be considered in determining the acceptability of entering the Mode or other specified condition in the Applicability, and any corresponding risk management actions. In addition, elements of acceptable risk assessment and risk management actions are included in Section 11 of NUMARC 93-01 "Assessment of Risk Resulting from Performance of Maintenance Activities," as endorsed by RG 1.182, which addresses general guidance for conduct of the risk assessment, gives quantitative and qualitative guidelines for establishing risk management actions, and provides example risk management actions. These changes are consistent with the guidance established by the ISTSs in consideration of the DBNPS CLB and, in view of the above, are acceptable.

Category 10 – Deletion of SR Shutdown Performance Requirements

The CTSs require maintaining LCO equipment operable by conducting SRs in accordance with specified SR intervals. The changes of this category relate to deleting the requirement to perform certain SRs during shutdown conditions only. The TSs that specify shutdown conditions would be changed to specify a frequency only. The control of the unit conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC staff to be unnecessary as a TS restriction. As indicated in NRC Generic Letter (GL) 91-04, allowing this control is consistent with the vast majority of other TS surveillances that do not dictate unit conditions for the surveillance. These changes are consistent with the guidance established by the ISTSs in consideration of the DBNPS CLB and, in view of the above, are acceptable.

For the reasons presented above, the proposed less restrictive changes to the CTSs are acceptable because they will not adversely impact safe operation of the facility. The ITS requirements are consistent with the CLB, operating experience, and plant accident and transient analyses, and provide reasonable assurance that public health and safety will be protected.

Table L attached to this SE lists the less restrictive changes being made in the DBNPS ITS conversion. Table L, which is organized in ITS order by each L-type DOC to the CTSs, provides a summary description of the less restrictive change that was made, the CTS and ITS references, and a reference to the specific change type discussed above.

D. Removed Details

When requirements have been shown to give little or no safety benefit, their removal from the TSs may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups' comments on the ISTSs. The NRC staff reviewed generic relaxations contained in the ISTSs and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The DBNPS design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in the ISTSs and thus provide a basis for ITSs. All of the changes to the CTSs involving the removal of specific requirements and detailed information from individual specifications evaluated to be Types 1 through 5 as described below:

Type 1 - Removing Details of System Design and System Description, Including Design Limits

The design of the facility is required to be described in the UFSAR by 10 CFR 50.34. In addition, the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 require that plant design be documented in controlled procedures and drawings and maintained in accordance with an NRC-approved Quality Assurance Program Manual (QAPM). The regulation at 10 CFR 50.59 specifies controls for changing the facility as described in the UFSAR. The regulation at 10 CFR 50.54(a) specifies criteria for changing the QAPM. The Technical Requirements Manual (TRM) is a general reference in the UFSAR and changes to it are accordingly also subject to 10 CFR 50.59. The ITS Bases also contain descriptions of system design. ITS 5.5.13 specifies controls for changing the Bases. Removing details of system design is acceptable because the associated CTS requirements being retained without these details are adequate to ensure safe operation of the facility. In addition, retaining such details in TS is unnecessary to ensure proper control of changes. Cycle-specific design limits are contained in the core operating limits report (COLR) in accordance with GL 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 3, 1988. ITS Section 5.6, "Reporting Requirements," includes the programmatic requirements for the COLR. Therefore, it is acceptable to remove Type 1 details from the CTSs and place them in licensee-controlled documents.

Type 2 - Removing Descriptions of System Operation

The plans for normal and emergency operation of the facility are required to be described in the UFSAR by 10 CFR 50.34. ITS 5.4.1.a and 5.4.1.e will require written procedures to be established, implemented, and maintained for plant operating procedures recommended in Appendix A of RG 1.33, "Quality Assurance Program Requirements (Operation)," Rev. 2, dated February 1978, and in all programs specified in ITS Section 5.5, respectively. The ITS Bases also contain descriptions of system operation. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the UFSAR and TRM. ITS 5.5.13 specifies controls for changing the Bases. Removing details of system operation is acceptable because the associated CTS requirements being retained without these details are adequate to ensure safe operation of the facility. In addition, retaining such details in TS is unnecessary to ensure proper control of changes. Therefore, it is acceptable to remove Type 2 details from the CTSs and place them in licensee-controlled documents.

Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements

Details for performing TS SRs or for regulatory reporting are more appropriately specified in the plant procedures. Prescriptive procedural information in a TS requirement is unlikely to contain all procedural considerations necessary for the plant operators to comply with TSs and all regulatory reporting requirements, and referral to plant procedures is therefore required in any event. Changes to procedural details include those associated with limits retained in the ITSs. For example, Specification 5.4.1 requires that written procedures covering activities that include all programs specified in Specification 5.5 be established, implemented, and maintained. ITS 5.5.7, "Inservice Testing Program," requires a program to provide controls for inservice testing (IST) of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves. The program includes defining testing frequencies specified in the ASME Operation and Maintenance Standards and Codes (OM Codes), and applicable addenda. The CTSs also contain requirements to test specific components such as pumps and valves, and to establish IST of Quality Group A, B, and C pumps and valves performed in accordance with the requirements for ASME Code Class 1, 2 and 3 components specified in the ASME OM Codes and addenda, subject to the applicable provisions of 10 CFR 50.55a. Therefore, it is acceptable to remove Type 3 details from the CTSs and place them in licensee-controlled documents.

Type 4 - Removal of a LCO, a SR, or other TS Requirement to the TS Bases, TRM, UFSAR, Offsite Dose Calculation Manual (ODCM), QAPM, Pressure and Temperature Limits Report (PTLR), IST Program, or Inservice Inspection Program (IIP)

Certain CTS administrative requirements are redundant with respect to current regulations and thus are relocated to the UFSAR or other appropriate licensee-controlled documents, including the TRM, ODCM, QAPM, or IIP. The FPS allows licensees to relocate to licensee-controlled documents CTS requirements that do not meet any of the criteria for mandatory inclusion in the TSs. Changes to the facility or to procedures as described in the UFSAR are made in accordance with 10 CFR 50.59. Changes made in accordance with the provisions of other licensee-controlled documents are subject to the specific requirements of those documents. For example, 10 CFR 50.54(a) governs changes to the QAPM, and ITS 5.5.13 governs changes to the ITS Bases. Therefore, it is acceptable to remove Type 4 details from CTSs and place them in licensee-controlled documents.

Type 5 - Removal of Cycle-Specific Parameter Limits from the TSs to the COLR

Certain CTS requirements contain cycle-specific parameter limits that are redundantly specified in the COLR, and thus, are relocated to the licensee-controlled COLR. The FPS allows licensees to relocate to licensee-controlled documents CTS requirements that do not meet any of the criteria for mandatory inclusion in the TSs. Changes are made to the COLR in accordance with the provisions of ITS 5.6.3. Therefore, it is acceptable to remove Type 5 details from CTSs and place them in licensee-controlled documents.

Table LA attached to this SE lists the less restrictive removal of detail changes being made in the DBNPS ITS Conversion. Table LA is organized in ITS order by each LA-type DOC and includes the following:

1. The ITS/CTS number, followed by the DOC number, (e.g. LA01);
2. The reference numbers of the associated CTS requirements;
3. A summary description of the relocated details and requirements;
4. The name of the licensee-controlled document to contain the relocated details and requirements (location);
5. The regulation (or ITS Specification) for controlling future changes to relocated requirements (change control process); and
6. A characterization of the type of change.

The NRC staff has concluded that these types of detailed information and specific requirements do not need to be included in the ITSs to ensure the effectiveness of the ITSs to adequately protect the health and safety of the public. Accordingly, these requirements may be moved to one of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement:

- Bases controlled in accordance with ITS 5.5.13, "Technical Specifications Bases Control Program."
- UFSAR (which references the TRM) controlled by 10 CFR 50.59.
- Programmatic documents required by ITS Section 5.5 and controlled by ITS Section 5.4.
- IST Program and IIP controlled by 10 CFR 50.55a.
- ODCM controlled by ITS 5.5.1.
- COLR controlled by ITS 5.6.3.
- PTLR controlled by ITS 5.6.4.
- QAPM, referenced in the UFSAR, and controlled by 10 CFR Part 50, Appendix B, and 10 CFR 50.54(a).

To the extent that information has been relocated to licensee-controlled documents, such information is not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. Further, where such information is contained in LCOs and associated requirements in the CTSs, the NRC staff has concluded that they do not fall within any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii) and discussed in the FPS (see Section 2.0 of this SE). Accordingly, existing detailed information, such as generally described above, may be removed from the CTSs and not included in the ITSs.

E. Relocated Specifications

The FPS states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria (now contained in 10 CFR 50.36(c)(2)(ii)) may be relocated from existing TSs (an NRC-controlled document) to appropriate licensee-controlled documents as noted in Section D above.

This section discusses the relocation of entire specifications from the CTSs to licensee-controlled documents. These specifications generally would include LCOs, Action Statements (i.e., Actions), and associated SRs. In its application and supplements, the licensee proposed relocating such specifications from the CTSs to a licensee-controlled document such as the TRM. The NRC staff has reviewed the licensee's submittals and finds that relocation of these requirements is acceptable in that the LCOs and associated requirements were found not to fall within the scope of 10 CFR 50.36(c)(2)(ii) and changes to licensee-controlled documents will be adequately controlled by 10 CFR 50.59, as applicable. These provisions will continue to be implemented by appropriate station procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures).

Table R attached to this SE lists the relocated changes that would be made in the DBNPS ITS conversion and lists all specifications that are being relocated from the CTSs to licensee-controlled documents. Table R includes the following in columns:

1. References to the ITS/CTS section and DOC number;
2. References to the relocated CTS requirement;
3. Summary descriptions of the relocated CTS requirement;
4. Names of the document that will contain the relocated specifications (i.e., the new location); and
5. The method for controlling future changes to the relocated specifications (i.e., the regulatory change control process).

The specifications relocated from the CTSs are not required to be in the TSs because they do not fall within the criteria for mandatory inclusion in the TSs as stated in 10 CFR 50.36(c)(2)(ii). These specifications are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. The NRC staff concludes that appropriate controls have been established for all of the current specifications and information being moved to the TRM. These relocations are the subject of a new license condition discussed in Section 7.0 of this SE. Until incorporated in licensee-controlled documents, changes to these specifications and information will be controlled in accordance with the current applicable procedures and regulations.

F. <u>Control of Specifications, Requirements, and Information Relocated from the CTS</u>

In the ITS Conversion, the licensee proposes to relocate specifications, requirements, and detailed information from the CTSs to licensee-controlled documents. This is discussed in Sections 4.D and 4.E of this SE. The facility and procedures described in the UFSAR and TRM can be revised in accordance with the provisions of 10 CFR 50.59, to ensure that records are maintained and appropriate controls are established over those requirements removed from the CTSs and future changes to the requirements. Other licensee-controlled documents contain provisions for making changes consistent with applicable regulatory requirements. For example, the ODCM can be changed only in accordance with ITS 5.5.1, and the administrative instructions that implement the QAPM can be changed in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the QAPM and such applicable regulations as 10 CFR 50.59.

The license condition for the relocation of requirements from the CTSs, which is discussed in Section 7.0 of this SE, will address the implementation of the ITS conversion and the schedule for the relocation of the CTS requirements into licensee-controlled documents.

G. Evaluation of Other TS Changes (Beyond-Scope Issues) Included in the Application for Conversion to ITS

This section evaluates other TS changes included in the licensee's ITS Conversion application. These changes include items that deviate from both the CTSs and the STSs. These changes are termed beyond-scope issues (BSIs). They were either identified by the licensee in its ITS application, or by the NRC staff during the course of its review. The BSIs were included in the Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for a Hearing published in the *Federal Register* on May 22, 2008 (73 FR 29787).

The following BSIs, listed below, do not have a corresponding SE due to the fact that the licensee has either chosen to keep their CTS or has decided to fully adopt the STS:

- BSI-11
- BSI-12
- BSI-14
- BSI-15
- BSI-16
- BSI-17
- BSI-18
- BSI-20
- BSI-23
- BSI-24

This section of the SE is divided into BSIs identified by the licensee (Section G.1) and those identified by the NRC staff (Section G.2).

G.1 BSI Changes Identified by the Licensee

G.1.1 BSI-1: ITS 3.3.8, DOC L03

BSI-1 proposes a change to the CTS by not requiring a CHANNEL CHECK of 2 relays (ITS 3.3.8, DOC L03). CTS 4.3-2 Functional Unit 4.b requires a CHANNEL CHECK of the Essential Bus Feeder Breaker Trip Degraded Voltage Relay (DVR) and Functional Unit 4.c requires a CHANNEL CHECK of the Diesel Generator Start and Load Shed on Essential Bus, Loss of Voltage Relay (LVR). ITS 3.3.8 does not require a CHANNEL CHECK.

G.1.1.1 Regulatory Evaluation

The NRC staff considered the following regulatory requirements and guidance in its review of the application:

Part 50 of 10 CFR, "Domestic Licensing of Production and Utilization Facilities," establishes the fundamental regulatory requirements. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

General Design Criterion (GDC) 10, "Reactor Design," in Appendix A to 10 CFR Part 50, requires that "the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

GDC 13, "Instrumentation and Control," requires that "instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."

GDC 20, "Protection System Functions," requires that the protection system "be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and initiate the operation of systems and components important to safety."

The regulation at 10 CFR 50.36, "Technical Specifications," states, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed TSs in accordance with the requirements of this section." Specifically, 10 CFR 50.36(c)(3) includes SRs "relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within [SLs], and that [LCOs] will be met."

G.1.1.2 Technical Evaluation

The licensee proposed a TS change to LCO 3.3.8 to delete the requirement for channel check surveillance every 12 hours for the loss of voltage and degraded voltage instrumentation for the emergency diesel generator (EDG) loss of power start (LOPS) function. The NRC has classified the issue related to this information as BSI-1. The NRC staff asked a question related to this BSI, and the licensee provided a response; both appear on the NRC/Davis-Besse ITS conversion website. The Instrumentation and Controls Branch (EICB) reviewed the licensee response for BSI-1.

The current TS 4.3-2 regarding Functional Units 4.b and 4.c requires the 12-hour channel check for the LVR and DVR for the EDG LOPS function. The licensee has requested to delete this surveillance test from ITS LCO 3.3.8. These safety functions are performed by voltage relays, which could be electromechanical or solid-state design. The licensee based this change on the fact that the channel check as described in ITS Section 1.0 uses observation to qualitatively assess channel behavior during operation. It should include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameters. However, this function is provided by voltage relays that do not provide any indication of voltage to the operators. The only indications the relays provide are to alarms, if they are tripped, in the control room or to a local target indicator that basically shows whether the channel is tripped. However, the relays do not indicate the value of the voltage at which the channel has tripped. Thus, the channel check requirement provides no qualitative information as to what voltage each relay is actually sensing and thus does not provide the status of each channel compared to the other channels, according to the licensee. In addition, the operator routinely monitors the status of the alarms in the control room. Therefore, the licensee concluded that it is not necessary to specify a channel check for these instruments. Based on these arguments, the licensee concluded that without the channel check, the loss of power instrumentation will continue to be tested in a manner and at a frequency necessary, to give confidence that the assumptions in the safety analysis will be met.

The NRC staff asked the licensee to provide details regarding these relays, including model and make information and the vendor manual. The licensee provided that information. Based on its review of the vendor manual, the NRC staff came to the same conclusion as the licensee, that without the channel check, the loss of power instrumentation will continue to be tested in a manner, to give confidence that the assumptions in the safety analysis will be met. Therefore, the NRC staff finds the proposed change acceptable.

G.1.1.3 Conclusion

The NRC staff reviewed BSI-1 related to a TS change in the ITS conversion of the DBNPS. Based on its review of the licensee's submittal and response to the NRC staff's question, the NRC staff finds that the proposed TS change related to BSI-1 is acceptable.

G 1.2 BSI-2: ITS 3.3.11, DOC M02

BSI-2 proposes a change to the CTS by changing the allowable values (AVs) for three functional units (ITS 3.3.11, DOC M02). CTS Table 3.3-12 Functional Unit 1, Steam Line Pressure-Low, specifies an AV of ≥ 591.6 per square inch gauge (psig) for the CHANNEL FUNCTIONAL TEST and ≥ 586.6 psig for CHANNEL CALIBRATION. CTS Table 3.3-12 Functional Unit 2, SG Level-Low, specifies an AV of ≥ 16.9 inches for the CHANNEL FUNCTIONAL TEST. CTS Table 3.3-12 Functional Unit 3, SG Feedwater Differential Pressure-High, specifies an AV of ≤ 197.6 psid (per square inch differential) for the CHANNEL FUNCTIONAL TEST and ≤ 199.6 psid for CHANNEL CALIBRATION. ITS Table 3.3.11-1 Functions 1, 3, and 2 specify AVs of ≥ 600.2 psig, ≥ 17.3 inches, and ≤ 176.8 psid, respectively.

G.1.2.1 Regulatory Evaluation

The NRC staff considered the following regulatory requirements and guidance in its review of the application:

In 10 CFR 50.36, the Commission established its regulatory requirements related to the contents of the TSs. According to 10 CFR 50.36, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section." Specifically, 10 CFR 50.36(c)(2)(i) defines limiting conditions for operation as "the lowest functional capability or performance levels of equipment required for safe operation of the facility." Furthermore, 10 CFR 50.36(c)(1)(ii)(A) states, "Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." The criteria for evaluating items to determine if a LCO of a nuclear reactor must be established appear in 10 CFR 50.36(c)(2)(ii). In addition, 10 CFR 50.36(c)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within [SLs], and that the [LCOs] will be met." The NRC staff reviewed the proposed TS changes against these requirements in 10 CFR 50.36 to ensure that there is reasonable assurance that the systems affected by the proposed TS changes will perform their required safety functions.

GDC 13, "Instrumentation and Control," in Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that the instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges during normal operation, anticipated operational occurrences (AOOs), and accident conditions. The NRC staff specifically reviewed the proposed TS changes and the affected instrument setpoint calculations and plant surveillance procedures to ensure proper operation of the steam and feedwater rupture control system (SFRCS).

GDC 15, "Reactor Coolant System Design," as it relates to 'the reactor coolant system (RCS) and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including [AOOs]."

GDC 20, "Protection System Functions," requires that the protection system be designed to initiate operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded and to initiate the operation of systems and components important to safety. The NRC staff established that the proposed TS change will assure that the fuel design limits and plant SLs specified in SFRCS TS 2.0 will not be exceeded with the proposed TS changes and will not affect the ability to initiate those systems and components important to safety.

RG 1.105, Rev. 3, "Setpoints for Safety-Related Instrumentations," issued December 1999, describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. The RG endorses Part I of ISA-S67.04-1994, "Setpoints for Nuclear Safety Instrumentation," subject to the NRC staff's clarifications. The NRC staff used this guide to establish the adequacy of the DBNPS setpoint calculation methodologies and the related plant surveillance procedures.

Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings [LSSSs] During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006, addresses the NRC's requirements on LSSSs assessed during periodic testing and calibration of instrumentation. RIS 2006-17 discusses issues that could occur during testing of LSSSs and that, therefore, may adversely affect equipment operability.

G.1.2.2 Technical Evaluation

The licensee proposed changes to the CTS Table 3.3-12 to modify trip setpoint AVs for the Steam Line Pressure-Low, SG Level-Low, and SG Feedwater Differential Pressure-High functional units instrumentation. These functions are moved to ITS Table 3.3-11. The SFRCS is designed to automatically start the Auxiliary Feedwater (AFW) system in the event of a main steam line break (MSLB), main feedwater (MFW) line rupture, a low level in the SGs or a loss of all four reactor coolant pumps. SFRCS is designed to automatically isolate the main steam system and MFW system in the event of MSLB or MFW line rupture and align to feed the unaffected SG upon a loss of steam pressure in one of the SGs. The NRC has classified the issue related to this information as BSI-2. The NRC staff asked questions related to this BSI, and the licensee responded to these questions; both the questions and the responses appear on the NRC / DBNPS ITS Conversion website.

The SFRCS is required to ensure an adequate FW supply to remove reactor decay heat during periods when normal FW supply has been lost. The licensee proposes to change the ITS Table 3.3.11-1 Functions 1 (Steam Line Pressure-Low), 3 (SG Level-Low), and 2 (SG Feedwater Differential Pressure-High) to ≥ 600.2 psig, ≥ 17.3 inches, and ≤ 176.8 psid, respectively. The proposed AVs were calculated using Methods 1 or 2 defined in ISA RP 67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." This change in the ITS deviates from the CTS and STS by only specifying a single AV and not two AVs, one applicable for a channel functional test and the other applicable for a channel calibration.

In its questions, posted on the website on December 26, 2007, the NRC asked the licensee to submit its setpoint methodology and to describe how it meets the NRC staff's guidance provided in RIS 2006-17. The licensee posted its response to these questions on the website on March 10, 2008.

The licensee submitted three calculations for these three functions showing the derivation of different values for the affected instrumentation. The licensee's calculations documented how the AVs, acceptable as-found tolerance, acceptable as-left tolerance limiting trip setpoint, and nominal trip setpoint are calculated from the analytical limit. The NRC staff reviewed calculation C-ICE-083.03-004 for the SG FW Differential Pressure-High function and determined that this calculation properly calculates all parameters in accordance with the guidance provided in RG 1.105, Rev. 3, and RIS 2006-17, except the acceptable as-found tolerance value. Based on this finding, the NRC staff posted more RAIs on the website on March 21, 2008. The licensee posted its responses on the website on April 1, and April 4, 2008. In its responses, the licensee agreed with the NRC staff's observation that the drift value being used in the calculation would mask the operability of the instrument and agreed to revise the calculation before implementing the ITS in a commitment. The licensee also stated that this change also applies to calculation C-ICE-083.03-003. The NRC staff reviewed the remaining two calculations and agreed that the comment also applies to calculation C-ICE-083.03-003 but not to calculation C-ICE-083.03-001. Based on this, the NRC staff has determined that the licensee's calculation meets the guidance in RG 1.105, Rev. 3, and RIS 2006-17.

In its response to additional RAIs, the licensee stated that these three setpoints are LSSSs that protect against violating SLs. Therefore, the licensee has proposed to add two notes to the channel functional test and channel calibration requirements in ITS Table 3.3.11-1 for these functions, consistent with similar notes in ITS 3.3.1 regarding the reactor protection system. In addition, the licensee has made changes to the bases, consistent with ITS 3.3.1. The licensee stated that these notes and bases changes are consistent with the guidance provided in RIS 2006-17.

The NRC staff requested the licensee to explain if this change was the result of a proposed power uprate and to justify the proposed AVs. The licensee explained that the proposed parameters were not due to the proposed uprate and that revised calculations were required for various reasons as determined by the DBNPS corrective action process. Also, the licensee has determined that these three functions are not listed as SL related LSSSs as required by 10 CFR 50.36(c)(ii)(A). The proposed parameters require the Functions to trip sooner than the AVs that are specified in the CTS, which is more conservative. The NRC staff agrees with the licensee's conclusions.

Based on the above discussion, the NRC staff finds the proposed changes to the TS acceptable.

G.1.2.3 Conclusion

The NRC staff has reviewed BSI-2 related to TS changes in the DBNPS ITS Conversion. Based on its review of the licensee's submittal and responses to the RAIs, the NRC staff finds that the proposed TS changes related to BSI-2 are acceptable.

G.1.3 BSI-3: ITS 3.4.1, DOC M01

BSI-3 proposes a change to the CTS by increasing the departure from nucleate boiling reactor coolant pressure parameter limits (ITS 3.4.1, DOC M01). CTS Table 3.2.-2 requires measured RCS pressure to be ≥ 2062.7 psig for four reactor coolant pump operation and ≥ 2058.7 psig for three reactor coolant pump operation. ITS LCO 3.4.1 requires RCS loop pressure to be ≥ 2064.8 psig for four reactor coolant pump operation and ≥ 2060.8 psig for three reactor coolant pump operation.

G.1.3.1 Regulatory Evaluation

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of SSCs that are required to protect the health and safety of the public. The Commission's regulatory requirements that are related to the content of the TSs are contained in 10 CFR 50.36, "Technical Specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) SLs, LSSSs and control settings; (2) LCO; (3) SR; (4) design features; and (5) administrative controls.

The NRC staff also applied the following regulatory requirement in reviewing the application:

GDC 15, as it relates to the RCS and associated auxiliary, control, and protection systems being "designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

G.1.3.2 Technical Evaluation

The licensee proposes to change the Reactor Coolant Pressure parameters for three and four reactor coolant pumps (RCPs) operating which relates to the departure from nucleate boiling (DNB) margin. The limits placed on DNB-related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed. The minimum RCS pressure is consistent with operation within the nominal operating envelop and corresponds to the initial pressure in the analyses. A pressure greater than the minimum pressure specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the plant to approach the DNB limit.

The licensee proposes to change the pressure for three and four RCPs operating to ≥ 2060.8 psig and ≥ 2064.8 psig, respectively. The CTS requires the measured reactor coolant pressure to be ≥ 2058.7 psig for three pumps operating and ≥ 2062.7 psig for four pumps operating. The proposed limits are consistent with the UFSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the minimum allowable DNBR throughout each analyzed transient.

The NRC staff requested that the licensee explain how the increase in the minimum pressure criterion will affect the emergency core cooling system (ECCS) capabilities and response times in thread 200712261240. Also, the NRC staff requested that the licensee explain how this change would affect the current loss-of-coolant accident (LOCA) analyses. The licensee stated "that the minimum pressure criterion is based on the minimum pressure drop from the core outlet to the hot-leg pressure tap. The fuel vendor previously identified that the calculated minimum pressure drop from the core outlet to the hot-leg pressure tap, upon which the CTS Table 3.2-2 minimum pressure criterion is based, was not correctly factored into the minimum pressure criterion. Therefore, the CTS reactor coolant pressure parameters listed in CTS Table 3.2-2 are slightly non-conservative. In order to offset this slight non-conservatism, a DNB penalty has been assessed in the past against the retained DNB margin in the reload licensing analyses. Once the proposed changes are made, this offset will no longer be necessary for future core reload analyses...No change is being made to the ECCS performance capabilities. The ECCS systems will continue to inject water at a flow rate that will provide adequate protection to the fuel and remove excessive heat. There is no change to the ECCS response time." The NRC staff has determined that these new parameters are more conservative than the previous parameters, therefore the NRC staff finds the proposed changes acceptable.

G.1.3.3 Conclusion

The NRC staff has reviewed BSI-3 related to TS changes in the DBNPS ITS Conversion. Based on its review of the licensee's submittal and responses to the RAIs, the NRC staff finds that the proposed TS changes related to BSI-3 are acceptable.

G.1.4 BSI-4: ITS 3.4.4, DOC L01

BSI-4 proposes a change to the CTS by extending the Completion Time to reduce the trip setpoints from "4 hours" to "10 hours" (ITS 3.4.4., DOC L01). CTS 3.4.1.1 Action A, requires a reduction of the High Flux trip setpoint from the four RCPs operating to three RCPs operating trip setpoint within 4 hours when shifting from four RCPs operating to three RCPs operating. ITS 3.4.4 Action A requires the reduction in the trip setpoints within 10 hours.

G.1.4.1 Regulatory Evaluation

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of structures, systems and components that are required to protect the health and safety of the public. The Commission's regulatory requirements that are related to the content of the TSs are contained in 10 CFR 50.36, "Technical Specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) SLs, LSSSs and control settings; (2) LCO; (3) SR; (4) design features; and (5) administrative controls.

The NRC staff also applied the following regulatory requirement in reviewing the application:

GDC 15, as it relates to the RCS and associated auxiliary, control, and protection systems being "designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences."

G.1.4.2 Technical Evaluation

The licensee proposes to increase the Completion Time to reduce the High Flux trip setpoint from four RCPs to three RCPs operating. The CTS, which applies when shifting from four RCPs operating to three RCPs operating, requires a reduction of the High Flux trip setpoint from the four RCPs operating to three RCPs operating trip setpoint within 4 hours. The ITS proposes to increase the Completion Time to 10 hours.

The STS is written for a plant whose design includes an automatic setdown feature for the nuclear overpower trip setpoint. That is, when shifting from four RCP operation to three RCP operation, the trip setpoints for the RPS instrumentation automatically adjust based on RCP configuration. The DBNPS design does not include this automatic setdown feature for the High Flux trip setpoints. The setpoints must be manually adjusted.

This change is similar to BSI-9, ITS 3.2.5, "Power Peaking Factors," to increase the Completion Time to 10 hours to reduce the High Flux and Flux-ΔFlux-Flow trip setpoints when F_Q or $F^N_{\Delta H}$ exceeds its limit in order to maintain both core protection and operability margin at the reduced thermal power. The Completion Time in ITS 3.4.4 has been increased to 10 hours to stay consistent with ITS 3.2.5 and provides reasonable time for repairs or replacement. ITS 3.2.5 Completion Time has been increased from 8 hours to 10 hours to be consistent with Completion Times for similar actions in STS 3.2.4 Required Actions A.1.2.2 and C.2. The NRC staff agrees with the licensee that the increase in time is reasonable based on the low probability of an accident occurring while operating outside the three RCPs operating trip setpoints, the automatic protection provided by the RPS and Flux-ΔFlux-Flow Function (which is automatically reset), the number of steps required to complete the Required Action 3.4.4 A.1, and the thermal power restriction provided in the LCO 3.4.4 b.1.

The NRC staff requested that the licensee provide more technical justification for increasing the Completion Time and provide the procedure for manually shifting from four RCP operation to three RCP operation in thread 200712030914. The licensee stated that "the procedure to reduce the high flux trip setpoints is performed on all 4 RPS channels. From a basic overview, the procedure for any one channel is: (1) Place associated Anticipatory Reactor Trip System

(ARTS) channel in bypass; (2) Place RPS channel in bypass; (3) Determine the setpoint voltage value that is equivalent to the three RCP AV; (4) The setpoint on the High Flux Trip bistable is adjusted (calibrated) to the lower required setpoint voltage; (5) A Functional Test is performed to make sure that the High Flux function trips within the required setpoint value; and (6) Restore the ARTS and RPS Channel." The NRC staff also evaluated ITS Table 3.3.1-1, "Reactor Protection System Instrumentation" Functions to ensure that they provided the same level of protection as the STS Table 3.3.1-1, "Reactor Protection System Instrumentation" Functions. The NRC staff found the two tables to be consistent and provide the same level of protection. Therefore, the NRC staff finds the proposed changes are acceptable.

G.1.4.3 Conclusion

The NRC staff has reviewed BSI-4 related to TS changes in the DBNPS ITS Conversion. Based on its review of the licensee's submittal and responses to the RAIs, the NRC staff finds that the proposed TS changes related to BSI-4 are acceptable.

G.1.5 BSI-5: ITS 3.5.1, DOC M01

BSI-5 proposes a change to the CTS by specifying a narrower range for the core flooding tank (CFT) borated water volume and nitrogen cover pressure (ITS 3.5.1, DOC M01). CTS LCO 3.5.1.b requires each CFT contained water volume be between 7555 gallons and 8004 gallons of borated water. CTS LCO 3.5.1.d requires each CFT nitrogen cover pressure be between 575 psig and 625 psig. In the ITS, SR 3.5.1.2 requires the borated water volume to be between 12.6 ft and 13.3 ft and ITS SR 3.5.1.3 requires the nitrogen cover pressure be between 580 psig and 620 psig.

G.1.5.1 Regulatory Evaluation

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of SSCs that are required to protect the health and safety of the public. The Commission's regulatory requirements that are related to the content of the TSs are contained in 10 CFR 50.36, "Technical Specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) SLs, LSSSs and control settings; (2) LCO; (3) SR; (4) design features; and (5) administrative controls.

The NRC staff also applied the following regulatory requirements in reviewing the application:

GDC 15, as it relates to the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB, are not exceeded during any condition of normal operation, including AOOs.

GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded during AOOs and that the core is cooled during accident conditions.

GDCs 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.

The regulation at 10 CFR 50.46 acceptance criteria for ECCS states that "each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an ECCS that must be designed so that its calculated cooling performance following postulated [LOCAs] conforms to the criteria set forth in" 10 CFR 50.46b.

G.1.5.2 Technical Evaluation

The licensee proposes to change the TS limits for the contained water volume and the nitrogen cover pressure of the CFTs. The CFTs supply water to the reactor during blowdown phase of a LOCA, to provide inventory to help accomplish the refill phase that follows thereafter, and to provide RCS makeup for a small-break LOCA. Two CFTs are provided for these functions. The CFTs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The CFTs are passive components, since no operator or control actions are required for them to perform their function. Internal tank pressure is sufficient to discharge the contents of the CFTs to the RCS if RCS pressure decreases below the CFT pressure.

The CTS require the contained water volume to be between 7555 gallons and 8004 gallons of borated water and the nitrogen cover pressure should be between 575 psig and 625 psig as per CTS 3.5.1. The ITS proposed limits for the contained water volume are required to be maintained between 12.6 feet and 13.3 feet and the nitrogen cover pressure is required to be maintained between 580 psig and 620 psig. This changes the CTS by specifying a narrower range for the CFT borated water volume and nitrogen cover pressure.

The licensee explained that the CFT borated water volume and nitrogen cover gas requirements specified in the CTS have not changed since the original issuance of the TS and are believed to be based on values that account for some instrument uncertainty. The licensee provided a Condition Report which included uncertainty calculations for the CFTs volume and pressure. Based on these calculations, it was identified by the licensee that surveillance acceptance criteria that were developed in this calculation warranted additional instrument uncertainty, which made the proposed limits more restrictive than the CTS 3.5.1 requirements for the CFTs contained volume and cover pressure.

The proposed limits indicated CFT water level limits requiring ≥ 12.6 feet and ≤ 13.3 feet are acceptable. These levels, corrected for instrument uncertainty, assure that the actual water volumes contained in the CFTs will remain between the analytical limits of 7480 gallons and 8078 gallons based on 1040 cu. Ft. -/+ 40. The proposed indicated CFT cover pressure limits requiring ≥ 580 psig and ≤ 620 psig have been found acceptable by the NRC staff. These pressures, corrected for instrument uncertainty, assure that the actual cover pressure in the CFTs will protect the analytical limit of 600 psig -/+ 33 psi. In the case of the CFT volume, the new value is also specified in feet, which is the readout of the available indication. After reviewing the provided information, the NRC staff finds the proposed changes acceptable.

G.1.5.3 Conclusion

The NRC staff has reviewed BSI-5 related to TS changes in the DBNPS ITS Conversion. Based on its review of the licensee's submittal and responses to the RAIs, the NRC staff finds that the proposed TS changes related to BSI-5 are acceptable.

G.1.6 BSI-10: ITS 3.1.9, DOC L03

BSI-10 proposes a change to the CTS by allowing the suspension of the RCS minimum temperature for criticality limit during performance of a MODE 2 PHYSICS TEST (ITS 3.1.9, DOC L03). However, it places a limitation on the RCS lowest loop average temperature that is allowed during the test. CTS 3.10.2 states that limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS. ITS 3.1.9 provides an additional exception to LCO 3.4.2, "RCS Minimum Temperature for Criticality," provided the RCS lowest loop average temperature is ≥ 520°F (ITS LCO 3.1.9 part e). A Surveillance to verify RCS lowest loop average temperature is ≥ 520°F every 30 minutes (ITS SR 3.1.9.2) has been added. In addition, ITS 3.1.9 ACTION C has been added to cover the situation when RCS lowest loop average temperature is not within limit. The Required Action is to suspend PHYSICS TESTS exceptions within 30 minutes. BSI-10 is out of sequence because it was identified by the licensee after the initial submittal.

G.1.6.1 Regulatory Evaluation

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of structures, systems and components that are required to protect the health and safety of the public. The Commission's regulatory requirements that are related to the content of the TSs are contained in 10 CFR 50.36, "Technical Specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) SLs, LSSSs and control settings; (2) LCO; (3) SR; (4) design features; and (5) administrative controls.
The NRC staff also applied the following regulatory requirement in reviewing the application:

GDC 15, as it relates to the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

G.1.6.2 Technical Evaluation

The licensee proposes to add a suspension during low power physics testing. The purpose of this MODE 2 LCO is to permit physics tests to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by 10 CFR Part 50, Appendix B. The licensee proposes to add suspension of LCO 3.4.2, "RCS Minimum Temperature for Criticality" provided that the "RCS lowest loop average temperature is ≥ 520 F." The purpose of LCO 3.4.2 is to prevent criticality outside the normal operating regime and to prevent operation in an unanalyzed condition. The licensee used guidance from the STS to add the suspension of LCO 3.4.2 and also TSTF-467 to add the RCS lowest loop average temperature requirement. Even though TSTF-467 is mentioned in the licensee's submittal, this is not an approved NRC document. TSTF-467 will not be referenced in the ITS.

The NRC staff requested that the licensee evaluate the effect, if any, that adding the RCS lowest loop average temperature requirement would have upon the minimum shutdown margin (SDM), particularly with respect to the no-load steam line break analysis. The licensee stated that the "ISTS 3.4.9 (Volume 6, Page 208) allows LCO 3.4.2, "RCS Minimum Temperature for Criticality," to be suspended during performance of a MODE 2 Physics Test. The ISTS Bases, Applicable Safety Analyses section (Page 214) (which has been maintained in the DBNPS ITS Bases) explains that: "Shutdown capability is preserved by limiting maximum obtainable THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the RCS temperature must be within the narrow range instrumentation for plant control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience." The Applicable Safety Analyses section of the Bases for STS 3.4.2 (Volume 9, Page 35) states that there are no accident analyses that dictate the minimum temperature for criticality. Furthermore, the STS 3.4.2 Bases Background section states that the reactor coolant moderator temperature coefficient used in core operating and accident analysis are defined for the normal operating temperature range. It also states that safety and operating analyses for lower temperatures have not been made. [DBNPS] has maintained the above information in the ITS Bases (it has all been placed in the Applicable Safety Analyses section), and has also included the following information: Compliance with the LCO ensures that the reactor will not be made or maintained critical at a temperature significantly less than the hot zero power (HZP) temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis. Therefore, as shown above, the STS Bases acknowledges that there are no safety analyses that assume a minimum temperature for criticality (MTC). The allowance to go below the normal limit in LCO 3.4.2 (525°F) is acceptable, as stated in STS 3.1.9 Bases, based on the low probability of an accident occurring and on prior operating experience. Thus, DBNPS does not believe that any special evaluation is required to adopt the allowance to go below the 525°F MTC limit of LCO 3.4.2, since the STS does not base the allowance on any special evaluation."

The licensee also stated that, "the STS 3.1.9 lower limit for the MTC was previously only stated in the Bases. TSTF-467T is correcting an error in the STS, in that the minimum limit must be specified in the TSs; it cannot only be specified in the Bases since the Bases cannot change the requirements of the TS (and STS 3.1.9, as written, specifically exempts the requirements of LCO 3.4.2)."

The licensee rightly states that there are no accident analyses that determine a MTC. However, there are accident analyses that are based upon certain values of SDM and MTC as initial conditions. SDM and MTC are related to core average temperature. The NRC staff agrees that the minimum RCS temperature may be allowed to decrease to 520°F, provided that SDM and MTC, which are key parameters in certain accident analyses (e.g., the no-load steam line break) are maintained such that the affected accident analyses of record remain valid. The NRC staff agrees that the minimum RCS temperature may be allowed to decrease below the normal limit in LCO 3.4.2 (525°F) as long as SDM and MTC values are maintained within the analyzed ranges. Therefore, the NRC staff finds the proposed changes acceptable.

G.1.6.3 Conclusion

Based on a review of the information that was provided and as discussed in the Technical Evaluation Section, the NRC staff has determined that the proposed changes are appropriate. The proposed changes are consistent with NRC practices and policies and therefore, the NRC staff has determined that the proposed changes should be approved.

The Commission has also concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

G.2 BSI Changes Identified by the NRC Staff

G.2.1 BSI-6: ITS 3.4.1, DOC L02

BSI-6 proposed a change to the CTS by delaying performance of a RCS flow Surveillance until adequate conditions exist to perform the Surveillance (ITS 3.4.1, DOC L02). CTS 4.2.5.2 requires the RCS total flow rate be determined to be within limits once per 18 months. ITS SR 3.4.1.4 requires the same Surveillance but includes a Note to allow the performance to be delayed for up to 7 days after stable thermal conditions are established at ≥ 70 percent rated thermal power (RTP).

G.2.1.1 Regulatory Evaluation

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of structures, systems and components that are required to protect the health and safety of the public. The Commission's regulatory requirements that are related to the content of the TSs are contained in 10 CFR 50.36, "Technical Specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) SLs, LSSSs and control settings; (2) LCO; (3) SR; (4) design features; and (5) administrative controls.

G.2.1.2 Technical Evaluation

The licensee proposed delaying the precision calorimetric heat balance SR to be performed until 7 days after stable thermal conditions are established at greater than or equal to 70 percent RTP. Babcock & Wilcox Owners Group (B&WOG) ISTS SR 3.4.1.4 NOTE - states that, "Only required to be performed when stable thermal conditions are established in the higher power range of MODE 1."

The purpose of this SR NOTE is to ensure the RCS total flow rate instrumentation is properly calibrated using a precision calorimetric heat balance. At lower power conditions, the thermal power is not stable and a precision calorimetric heat balance could not provide accurate results. The NRC staff requested the licensee to explain why they need 7 days to perform this SR when the plant is at 70 percent RTP and stable conditions were established. In response to the RAI, the licensee revised their proposal to state that the performance of the SR be delayed for up to 24 hours to perform the precision heat balance to allow for stable thermal conditions to be established at greater than or equal to 70 percent RTP. B&WOG ISTS SR 3.4.1.4 NOTE does not have any specific requirements. However, the revised proposal is consistent with Westinghouse Owners Group ISTS and Combustion Engineering Owners Group ISTS SR 3.4.1.4 NOTE. After reviewing the provided information, the NRC staff finds the revised proposal to perform this SR after 24 hours when the plant stable conditions are established at 70 percent RTP is acceptable. The licensee has reflected this change in the NOTE proposal is reflected in the supplement to this section of the ITS Conversion amendment.

G.2.1.3 Conclusion

Based on the NRC staff's review of the licensee's submittal and response to the RAI, the NRC staff finds that the proposed TS change related to BSI-6 is acceptable.

G.2.2 BSI-7: ITS 3.8.1, DOC M06

BSI-7 proposes a change to the CTS by requiring the EDGs to be tested for a longer duration, at higher loading, and within a power factor (PF) limit, with an allowance to not meet the load or PF requirements due to momentary transients (ITS 3.8.1, DOC M06). CTS 4.8.1.1.2.d.3 requires verification that the diesel generator (DG) operates for ≥ 60 minutes while loaded to ≥2000 kW. ITS SR 3.8.1.13 requires an endurance and load test for each EDG. The endurance and load test requires that the EDGs be operated for ≥ 8 hours, with ≥ 2 hours loaded between 2730 kW and 2860 kW and the remaining 6 hours loaded between 2340 kW and 2600 kW. This Surveillance is modified by Note 1 and Note 3. Note 1 states, "momentary transients outside the load and PF ranges do not invalidate this test." Note 3 states, "If performed with EDG synchronized with offsite power, it shall be performed within the PF limit. However, if grid conditions do not permit, the PF limit is not required to be met. Under this condition the PF shall be maintained as close to the limit as practicable."

G.2.2.1 Regulatory Evaluation

The NRC staff used the following NRC requirements and guidance documents to review the licensee's amendment request:

The regulation in 10 CFR Part 50 requires that TS shall be included by applicants for a license authorizing operation of a production or utilization facility. 10 CFR 50.36(c) requires that TS include items in five specific categories related to station operation. These categories are: (1) SLs, LSSSs, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The proposed change to the TS (BSI-7) is within the SR category.

Safety Guide 9, March 1971, "Selection of Diesel Generator Set Capacity for Standby Power Supplies" (superseded by NRC RG 1.9) describes an acceptable basis for the selection of DG sets of sufficient capacity and margin to implement General Design Criterion 17 of Appendix A to 10 CFR Part 50.

RG 1.108, August 1977, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power systems at Nuclear Power Plants" dated August 1977 (withdrawn), described a method acceptable to the NRC staff for complying with the Commission's regulations with regard to periodic testing of diesel electric power units. This RG has since been merged into the RG 1.9.

RG 1.9, Rev. 3, July 1993, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power systems at Nuclear Power Plants," describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to design and testing of DGs. This RG is superseded by RG 1.9, Rev. 4, March 2007, "Application and Testing of Safety-Related Diesel Generators at Nuclear Power Plants."

G.2.2.2 Technical Evaluation

According to the DBNPS UFSAR, the plant has two EDGs; each EDG has a continuous rating of 2600 kilowatts (kW) @ 0.8 PF and a short term rating of 2860 kW @ 0.8 PF. The short-term rating is defined as the electric power capability that the EDG can maintain in the service environment for 2 hours in any 24-hour period.

The CTS SR 4.8.1.1.2.d.3 states as follows:

> Verifying the diesel generator operated for ≥ 60 minutes while loaded to ≥ 2000 kW. This SR is performed once each REFUELING INTERVAL during shutdown.

The proposed ITS SR 3.8.1.13 would read as follows:

> Verify each EDG operates for ≥ 8 hours:
>
> a. For ≥ 2 hours loaded ≥ 2730 kW and ≤ 2860 kW [105 percent to 110 percent of the EDG continuous rating]; and
>
> b. For the remaining hours of the test loaded ≥ 2340 kW and ≤ 2600 kW [90 percent to 100 percent of the EDG continuous rating]
>
> The above test is to be performed every 24 months. The following notes are applicable to the above test:
>
> 1. Momentary transients outside the load and PF ranges do not invalidate this test.
>
> 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.
>
> 3. If performed with EDG synchronized with offsite power, it shall be performed within the PF limit. However, if grid conditions do not permit, the PF limit is not required to be met. Under this condition the PF shall be maintained as close to the limit as practicable.
>
> The PF limit is specified in the ITS Bases as follows: ≤ 0.90 for part 'a' of the test (when an EDG is tested at load equivalent to 105 percent to 110 percent of the EDG continuous rating); and ≤ 0.85 for EDG 1 and ≤ 0.86 for EDG 2 for part 'b' of the test (when an EDG is tested at load equivalent to 90 percent to 100 percent of the EDG

> continuous rating). In the TS Bases, it is stated that the PF is representative of the actual inductive loading an EDG would see under design-basis accident conditions.

When comparing the above ITS SR 3.8.1.13 to the corresponding STS SR 3.8.1.14, the NRC staff identified the following differences:

8 hours versus 24 hours Endurance Run

The CTS SR 4.8.1.1.2.d.3 requires the EDG load surveillance (endurance run) to be performed for a minimum of one hour at least once each refueling interval. The STS SR requires EDGs to be tested for 24 hours, while the proposed ITS SR would require the EDG to be tested for 8 hours. Thus, while the proposed ITS SR is more conservative than the CTS SR, it is less conservative with respect to the STS SR. RG 1.108 (basis for the STS SR) and RG 1.9 (Revisions 3 and 4) recommend a 24-hour EDG endurance run test.

In the LAR, the licensee provided the following justifications for the 8-hour EDG endurance run:

> The purpose of CTS 4.8.1.1.2.d.3 is to ensure the EDG can supply the emergency loads. This change requires the EDGs to be tested at a load range of 105 percent to 110 percent for 2 continuous hours and a load range of 90 percent to 100 percent within the power factor limit, if applicable, for 6 hours, consistent with the recommendations of Institute of Electrical and Electornics Engineers (IEEE) Standard 387-1995. This change is designated as more restrictive because it adds more stringent testing requirements to the CTS.

The NRC staff requested the licensee to provide the maximum design basis EDG loads to ensure the ITS SR endurance run will envelop the maximum design basis loads. In its response dated May 9, 2008, the licensee stated that the maximum expected accident load for EDG 1 is 2444 kW. The maximum expected accident load for EDG 2 is 2507 kW. Both load values assume the maximum frequency effect of 61.2 hertz (Hz). The licensee also provided an excerpt from the EDG loading calculation to confirm the above values. In this calculation, the Appendix R loading was shown as 102 percent of the EDG continuous rating. Based on RG 1.9, Rev. 3, the NRC staff has typically required that EDG accident loading be less than the continuous rating of EDG, so that the endurance run would provide reasonable assurance that the EDG will be able to supply the accident loads. The NRC staff also observed that the above Appendix R loading of 102 percent was corresponding to a frequency of 61.2 Hz. In a letter dated June 18, 2008, the NRC staff requested the licensee to explain the discrepancy between this value and the proposed ITS value of 60.5 Hz. In its response dated July 1, 2008, the licensee stated that the maximum Appendix R loading will be 2550 kW corresponding to a frequency of 60.5 Hz, which is less than the continuous rating of the EDG. Considering that the maximum calculated accident loading (including Appendix R loading) will be less than the continuous rating of EDG and the 8-hour endurance run will envelop the postulated loading, the NRC staff finds the EDG kW loading test in the proposed ITS SR 3.8.1.13 to be acceptable.

Based on RG 1.108 (basis for STS SR), and RG 1.9 (Revisions 3 and 4), the NRC staff has required licensees to perform the EDG endurance run for 24 hours. In Rev. 4 of RG 1.9, the NRC took exception to the endurance run of 8 hours as specified in IEEE 387-1995. Operating experience indicates that weaknesses in EDG systems can be identified during the 24 hour endurance run. Thus, the NRC staff finds that a 24-hour endurance run can help in the early identification of potential EDG failures and can improve the reliability of an EDG to meet its mission time when required for safe shutdown of the plant. However, the NRC staff also finds that the licensee proposed 8-hour endurance run test is more conservative than the present 1-hour endurance run test specified in the CTS. Therefore, the NRC staff concludes that the 8-hour EDG endurance run is acceptable.

PF Limit

Presently, CTS SR 4.8.1.1.2.d.3 does not have any requirement for PF testing of the EDGs. According to Note 3 of the ITS SR 3.8.1.13, the licensee has proposed to perform the PF limit test when the EDG is synchronized with offsite power and performing a load test. The proposed PF limit test requirement in the ITS is similar to the power test requirement in the STS except that the PF limit value is specified in the ITS Bases.

In the LAR, the licensee provided the following justification (JFD # 14) for PF related deviation from the STS:

> Currently, there are no power factor limit requirements in the CTS. The specific power factor value is included in the ITS Bases and will therefore be controlled under ITS 5.5.13, the TS Bases Control Program. This program provides for the evaluation of changes to ensure the Bases are properly controlled.

The NRC staff finds the proposed PF limit test acceptable because the test specified in the ITS is more conservative than the CTS, since the CTS do not provide PF limit test, and the proposed PF values specified in the TS Bases envelop the maximum design-basis accident loads. The PF limit values: ≤ 0.90 for part 'a' of the test (at load equivalent to 105 percent to 110 percent of the EDG continuous rating); and ≤ 0.85 for EDG 1 and ≤ 0.86 for EDG 2 for part 'b' of the test (at load equivalent to 90 percent to 100 percent of the EDG continuous rating), are also consistent with the intent of PF testing recommended by RG 1.9.

G.2.2.3 Conclusion

The NRC staff has reviewed the licensee's proposed ITS SR 3.8.1.13. Based on the above information, the NRC staff finds the proposed ITS SR 3.8.1.13 acceptable as it is more conservative than CTS SR 4.8.1.1.2.d.3. Furthermore, the proposed ITS SR 3.8.1.13 meets the regulatory intent of 10 CFR 50.36(c).

G.2.3 BSI-8: ITS 5.5.16, DOC A.6

BSI-8 proposes a change to incorporate TSTF-451T, "Correct the Battery Monitoring and Maintenance Program and the Bases of SR 3.8.4.2" (ITS 5.5.16, DOC A.6).

G.2.3.1 Regulatory Evaluation

The following NRC requirements and guidance documents are applicable to the NRC staff's review of the licensee's amendment request:

Part 50 of 10 CFR, Appendix A, GDC 17, "Electric power systems," requires, in part, that "an onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety ... The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions ... Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies."

GDC 18, "Inspection and testing of electric power systems," requires, in part, that "Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features ..."

Part 50.36(c)(2)(ii) of 10 CFR, "Technical Specifications," requires that "[a] technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the [criteria set forth in 10 CFR 50.36(c)(2)(ii)(A)-(D)]."

Part 50.36(c)(3) of 10 CFR, "Technical Specifications," requires that TSs include SRs, which "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

Part 50.63 of 10 CFR, "Loss of all alternating current [AC] power," requires, in part, that each light-water-cooled nuclear power plant licensed to operate "must be able to withstand for a specified duration and recover from a station blackout as defined in §50.2."

Part 50.65(a)(3) of 10 CFR, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires, in part, that "Performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle provided the interval between evaluations does not exceed 24 months ... Adjustments shall be made where necessary to ensure that the objective of preventing failures of structures, systems, and components through maintenance is appropriately balanced against the objective of minimizing unavailability of structures, systems, and components due to monitoring or preventive maintenance."

RG 1.32, Rev. 3, "Criteria for Power Systems for Nuclear Power Plants," provides guidance for complying with GDCs 17 and 18 with respect to the design, operation, and testing of safety-related electric power systems of all types of nuclear power plants.

RG 1.93, "Availability of Electric Power Sources," describes the operating procedures and restrictions acceptable to the staff which should be implemented if the available electric power sources are less than the LCO.

RG 1.129, Rev. 2, "Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants," provides guidance for complying with GDCs 1, 17, and 18 with respect to the maintenance, testing, and replacement of vented lead-acid storage batteries in nuclear power plants.

G.2.3.2 Technical Evaluation

G.2.3.2.1 ITS 3.8.6 Change (1)

The licensee proposed the following:

> Delete TS Table 4.8-1 and relocate the following limits to the Battery Monitoring and Maintenance Program specified in new TS Section 5.5.16:
>
> - Category A and B limits for cell voltage and electrolyte level.
>
> Replace verification requirements for battery cell specific gravity monitoring with float current monitoring requirements.

Evaluation of ITS 3.8.6 Change (1)

TS Table 4.8-1 specifies the battery cell parameter requirements, including electrolyte level, float voltage, and specific gravity. The Category A and B values of TS Table 4.8-1 represent appropriate monitoring levels and appropriate preventive maintenance levels for long-term battery quality and extended battery life. Paragraph 50.36(c)(2)(i) of 10 CFR states, in part "[LCOs] are the lowest functional capability or performance levels of equipment required for safe operation of the facility." As such, the Category A and B values for cell voltage and electrolyte level do not reflect the 10 CFR 50.36 criteria for LCOs. The licensee proposed relocating these parameters and the Required Actions associated with restoration to a licensee-controlled program.

In response to a NRC staff RAI, the licensee provided a Regulatory Commitment to maintain the existing surveillances for the battery parameters (i.e., visual inspection, cell-to-cell connection resistance, specific gravity, etc.) that are to be relocated to the new Battery Monitoring and Maintenance Program. Based on this information, the NRC staff has reasonable assurance that the battery parameter values will continue to be controlled at their current level, and that actions to restore deficient values will be implemented in accordance with the licensee's corrective action program. Furthermore, the battery and its preventive maintenance and monitoring program are under the regulatory requirements of 10 CFR 50.65. The relocation of the aforementioned battery parameters will continue to assure that the battery is maintained at current levels of performance, and that operators continue to monitor the battery parameters for degradation.

The licensee also proposed relocating the Category B specific limiting values of TS Table 4.8-1 for the battery electrolyte level and temperature to a licensee-controlled program (TS 5.5.16). However, new TS 3.8.6, Conditions C and D, will require the electrolyte temperature (pilot cell only) and level (any battery cell) to be greater than or equal to minimum established design limits. The licensee proposed relocating the electrolyte temperature and level criteria (i.e., the minimum established design limits) to the DBNPS TS Bases. Depending on the available excess capacity of the associated battery, the minimum temperature necessary to support operability of the battery can vary. Relocating these values to a licensee-controlled program will provide the licensee with added flexibility to monitor and control this limit at values directly related to the battery's ability to perform its assumed function. The NRC staff concludes that the Category B specific limiting values for TS Table 4.8-1 for the battery electrolyte level and temperature do not meet the criteria of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs and may be relocated to a licensee-controlled program. Therefore, the NRC staff finds that these changes are acceptable.

The licensee proposed replacing the requirements to measure battery cell specific gravity with requirements to monitor float current. In response to a RAI on this subject, the licensee provided a letter from its battery manufacturer (GNB Industrial Power), which concurred with the use of float current monitoring for the purpose of determining the state of charge of the DBNPS batteries. The licensee also provided a Regulatory Commitment to maintain a 5 percent recharge margin for each battery to ensure that the 2-amp float current value provides an indication that the battery is fully charged (i.e., fully capable of performing its design function). The licensee stated that a 95 percent charged battery represents a fully charged battery at DBNPS. In response to a RAI, the licensee indicated that the proposed 2-amp float current value equates to a 95 percent charged battery on the battery manufacturer's recharge curve. The licensee further noted that the battery manufacturer previously performed testing in support of a prior LAR that demonstrated that the 2-amp float current equates to a 95 percent charged battery.

The licensee stated that the equipment that will be used to monitor float current will have the necessary accuracy and capability to measure electrical currents in the expected range. The licensee stated that it has successfully performed testing that demonstrated the accuracy and capability of the equipment. Based on its review of the provided information, the NRC staff has concluded that the equipment for monitoring battery float current will have the necessary accuracy and capability to measure electrical currents in the expected range.

The NRC staff finds that the concurrence of the battery manufacturer, the Regulatory Commitment to maintain a 5 percent recharge margin, and the demonstrated accuracy and capability of the float monitoring equipment provides reasonable assurance that the deletion of the requirement to monitor specific gravity will not have a significant impact on safety or the ability to accurately determine the operability of the DBNPS batteries.

The proposed changes discussed above ensure the battery parameters (maintenance, testing, and monitoring) are performed in accordance with the "Battery Monitoring and Maintenance Program," as specified in TS Section 5.5.16. The NRC staff concludes that there is reasonable assurance that safe plant conditions will continue to be maintained; therefore, the proposed changes are acceptable.

G.2.3.2.2 ITS 3.8.6 Change (2)

The licensee proposed adding new TS 3.8.6. The new Conditions, with their associated Required Actions, provide compensatory actions for specific abnormal battery conditions, as follows:

> Condition A addresses the situation in which one or more batteries have one or more battery cells with a float voltage less than or equal to 2.07 volts (V).
>
> Condition B addresses the situation in which one or more batteries are found with a float current greater than 2 amps.

Condition C addresses the situation in which one or more batteries have one or more cells with electrolyte level less than the minimum established design limits.

Condition D addresses the situation in which one or more batteries are found with pilot cell electrolyte temperature less than minimum established design limits.

Condition E addresses the situation in which batteries in redundant trains are found with battery parameters not within limits.

Condition F addresses the situation in which Required Action and associated Completion Time (CT) of Condition A, B, C, D, or E are not met, OR one or more cells with float voltage less than or equal to 2.07 V and float current greater than 2 amps, OR SR 3.8.6.6 is not met.

The licensee also proposed adding the following SRs to TS 3.8.6:

SR 3.8.6.1 requires verification that float current for each battery is less than or equal to 2 amps every 7 days.

SR 3.8.6.2 requires verification that each battery pilot cell voltage is greater than 2.07 V every 31 days.

SR 3.8.6.3 requires verification that each battery connected cell electrolyte level is greater than or equal to minimum established design limits every 31 days.

SR 3.8.6.4 requires verification that each battery pilot cell temperature is greater than or equal to minimum established design limits every 31 days.
SR 3.8.6.5 requires verification that each battery connected cell voltage is greater than 2.07 V every 92 days.

SR 3.8.6.6 requires verification that the battery capacity is greater than or equal to 80 percent of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test every 60 months AND 12 months when the battery shows degradation, or has reached 85 percent of the expected life with capacity less than 100 percent of the manufacturer's rating, AND 24 months when battery has reached 85 percent of the expected life with capacity greater than or equal to 100 percent of the manufacturer's rating.

Evaluation of ITS 3.8.6 Change (2)

The licensee proposed adding new TS 3.8.6, Condition A to address what was formerly the Category B limit for float voltage in TS Table 4.8-1. This new Condition would be applicable when one or more batteries is found with one or more battery cells with a float voltage less than or equal to 2.07 V. Once Condition A has been entered, the battery cell is considered degraded and the Required Actions are to: (A.1) verify within 2 hours that the battery terminal voltage is greater than or equal to the minimum established float voltage (SR 3.8.4.1), (A.2) verify within 2 hours that each battery's float current is less than or equal to 2 amps (SR 3.8.6.1), and (A.3) restore affected cell voltage to greater than 2.07 V. The above actions ensure that the battery has adequate capacity to perform its intended function. Continued operation for up to 24 hours is proposed to allow the restoration of the affected cells' voltage to greater than or equal to 2.07 V. The NRC staff considers that the 24-hour restoration time is reasonable, that it maintains safe plant conditions and, therefore, is acceptable.

The licensee proposed adding new TS 3.8.6, Condition B to address battery state of charge. This new Condition would be applicable when one or more batteries are found with a float current greater than 2 amps. A float current of greater than 2 amps provides an indication that a partial discharge has occurred. The Required Actions are: (B.1) verify within 2 hours that the battery terminal voltage is greater than or equal to the minimum established float voltage (SR 3.8.4.1), thus confirming battery charger operability, and (B.2) restore battery float current to less than or equal to 2 amps. If the terminal voltage is satisfactory and there are no battery cells with a voltage less than 2.07 V, Required Action B.2 of Condition B assures that within 12 hours the battery will be restored to its fully-charged condition from any discharge that might have occurred due to a temporary loss of the battery charger.

If the terminal voltage is found to be less than the minimum established float voltage, it indicates that the battery charger is either inoperable or is operating in the current limit mode. If the battery charger is operating in the current limit mode for 2 hours, it is an indication that the battery has been substantially discharged and likely cannot perform its required design functions.

If the float voltage is found to be satisfactory, but there are one or more battery cells with a float voltage less than or equal to 2.07 V, the associated "OR" statement in the revised Condition F of TS 3.8.6 would be applicable, and the battery must immediately be declared inoperable. If float voltage is satisfactory and there are no cells less than or equal to 2.07 V, and the out-of-limit float current condition is due to one or more battery cells with low voltage, then the battery is not substantially discharged. For this condition, the NRC staff finds that the 12-hour CT to restore battery float current to within limits is reasonable. The NRC staff concludes that adding new TS 3.8.6, Condition B is reasonable, maintains safe plant conditions and, therefore, is acceptable.

The licensee proposed adding new TS 3.8.6, Condition C to address the electrolyte level in a cell. This new Condition would be applicable when one or more batteries are found with one or more cells with an electrolyte level less than the minimum established design limits. If the level is above the top of the battery cell plates, but below the minimum limit (i.e., minimum level indication mark on the battery cell jar), the battery still has sufficient capacity to perform its intended safety function and is considered operable. With the electrolyte level below the top of the plates, there is a potential for dry-out and plate degradation. New Required Actions C.1, C.2, and C.3 (as well as provisions in new TS 5.5.16) restore the electrolyte level, ensure that the cause of the loss of the electrolyte is not due to a leak in the battery cell jar, and equalize and test the affected battery. Based on its review of the provided information, the NRC staff concludes that these changes are adequate to ensure that minimum electrolyte levels are maintained and, therefore, are acceptable.

The licensee proposes to add new TS 3.8.6, Condition D which applies to a battery found with a pilot cell electrolyte temperature less than the minimum established design limit. This new Condition would be applicable when one or more batteries have a pilot cell electrolyte temperature less than minimum established design limits. A low electrolyte temperature limits the current and power available from the battery.

During its review, the NRC staff requested that the licensee provide assurance that a battery with a battery pilot cell electrolyte temperature slightly greater than or equal to the minimum established design limit will remain capable of performing its minimum design function. In responding to the RAI, the licensee stated that the 5°F temperature deviation criteria, as specified in industry guidance documents, cannot always be demonstrated. As a result, the licensee has provided a Regulatory Commitment to use individual cell temperature as one of the criteria for selecting pilot cells or a separate pilot cell will be selected to reflect average battery temperature. Based on this information, the NRC staff concludes that the pilot cell temperature will provide an accurate representation of the temperature of the battery bank. The 12-hour CT provides a reasonable time to restore the electrolyte temperature within established limits. The NRC staff concludes that the proposed change is adequate to ensure that the minimum electrolyte temperature is maintained and, therefore, is acceptable.

The licensee proposed adding new TS 3.8.6, Condition E to address the condition where two or more redundant train battery parameters are not within limits. If this condition exists, there is not sufficient assurance that the batteries will be capable of performing their intended safety function. With redundant batteries involved, multiple systems that rely on DC power could be affected. The licensee proposed that battery parameters for the affected battery in one train be restored to within limits within 2 hours. The NRC staff finds that the proposed change is reasonable, maintains safe plant conditions and, therefore, is acceptable based on its review of the information provided by the licensee.

The licensee proposed adding new TS 3.8.6, Condition F to provide a default condition for battery parameters that fall outside the allowance of the Required Actions for Condition A, B, C, D, or E. Under this condition, it is assumed that sufficient capacity is not available to supply the maximum expected load requirements. New Condition F also addresses the case where one or more batteries is found with one or more battery cells that have a float voltage less than or equal to 2.07 V and a float current greater than 2 amps. The NRC staff concludes, after reviewing the information provided, that the proposed change is reasonable, maintains safe plant conditions and, therefore, is acceptable.

The licensee proposed adding new SR 3.8.6.1, which will require verification that the float current for each battery is less than or equal to 2 amps every 7 days. The purpose of this SR is to determine the state of charge of the battery. Float charge is the condition in which the battery charger is supplying the continuous small amount of current (i.e., less than 2 amps) required to overcome the internal losses of a battery to maintain the battery in a fully charged state. The float current requirements are based on the float current indicative of a charged battery, as specified by the battery manufacturer. As stated in the evaluation of TS 3.8.6 change (1) above, the use of float current to determine the state of charge of the battery is consistent with DBNPS's battery manufacturer recommendations. The NRC staff concludes that this change is reasonable, maintains safe plant conditions and, therefore, is acceptable, based on its review of the information provided by the licensee and the battery manufacturers recommendations.

The licensee proposed adding new SR 3.8.6.2 and SR 3.8.6.5, which will require verification that the float voltage of pilot cells and all connected cells are greater than 2.07 V every 31 and 92 days, respectively. This voltage level represents the point where battery operability is in question. The Battery Monitoring and Maintenance Program (new TS Section 5.5.16) includes actions to restore battery cells with float voltage less than 2.13 V and actions to verify that the remaining cells are greater than 2.07 V when a cell or cells have been found to be less than

2.13 V. The NRC staff concludes that these changes are reasonable, maintain safe plant conditions and, therefore, are acceptable based upon the information provided by the licensee.

The licensee proposed adding SR 3.8.6.3, which will require verification that the electrolyte level of each connected cell of each battery is greater than or equal to the minimum established design limits every 31 days. Operation of the batteries at electrolyte levels greater than the minimum established design limit ensures that the battery plates do not suffer physical damage and continue to maintain adequate electron transfer capability. Upon review of the information provided by the licensee, the NRC staff concludes that this change will ensure that minimum electrolyte levels are maintained and, therefore, is acceptable.

The licensee proposed adding SR 3.8.6.4, which will require verification that the temperature of each battery pilot cell is greater than or equal to the minimum established design limits every 31 days. As mentioned previously, the licensee has provided a Regulatory Commitment to use cell temperature as one of the criteria for selecting pilot cells or a separate pilot cell will be selected to reflect average battery temperature. Based on this information, the NRC staff concludes that the pilot cell temperature will provide an accurate representation of the temperature of the battery bank and this change is therefore acceptable.

The licensee proposed relocating existing SR 4.8.2.3.2.e to SR 3.8.6.6. This SR will continue to require verification that the battery capacity is greater than or equal to 80 percent of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test (1) every 60 months, AND (2) 12 months when the battery shows degradation, or has reached 85 percent of the expected life with capacity less than 100 percent of the manufacturer's rating, AND (3) 24 months when battery has reached 85 percent of the expected life with capacity greater than or equal to 100 percent of the manufacturer's rating.

The NRC staff finds that this change is administrative in nature, and therefore, is acceptable.

G.2.3.2.3 ITS 3.8.6 Change (3)

The licensee proposed creating a new program, called the "Battery Monitoring and Maintenance Program," in new TS Section 5.5.16. This program will have elements relocated from the different affected TSs. The program will be specified in the TSs as follows:

5.5.16 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V;

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; and

c. Actions to verify that the remaining cells are > 2.07 V when a cell or cells have been found to be < 2.13 V.

Evaluation of ITS 3.8.6 Change (3)

The licensee proposed adding a new program, the Battery Monitoring and Maintenance Program, to be specified in new TS Section 5.5.16. The NRC staff understands that the licensee plans to use the recommendations provided in the IEEE Standard (Std.) 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," to develop the proposed battery monitoring and maintenance program prescribed by new TS 5.5.16. However, the staff would like to note that this version of IEEE Std. 450 has not been officially endorsed by the NRC.

As noted above, the licensee provided a Regulatory Commitment to include requirements for monitoring the following battery parameters and applicable acceptance criteria in the new Battery Monitoring and Maintenance Program: visual inspection (i.e., for corrosion), cell-to-cell connection resistance, and specific gravity.

Based on the above, the staff has reasonable assurance that the battery parameter values will continue to be controlled at their current level, and actions to restore deficient parameters will be implemented in accordance with the licensee's corrective action program. Furthermore, the battery and its preventive maintenance and monitoring program continue to be subject to the regulatory requirements of 10 CFR 50.65.

The staff concludes that this change will continue to assure the battery is maintained at current levels of performance, and appropriately focuses operators on the monitoring of battery parameter degradations and, therefore, is acceptable.

G.2.3 Conclusion

Based on the above evaluation and the Regulatory Commitments listed below, the staff finds the proposed revisions to the DBNPS TSs provide reasonable assurance of the continued availability of the required AC and DC power to shut down the reactor and to maintain the reactor in a safe condition after an anticipated operational occurrence or a postulated DBA. The staff also concludes that the proposed TS changes are in accordance with 10 CFR 50.36, 10 CFR 50.65, and the requirements of GDCs 17 and 18. Therefore, the NRC staff finds the proposed changes acceptable.

G.2.4 BSI-9: ITS 3.2.5, DOC L02

BSI-9 proposes a change to the CTS by extending the CT of the High Flux and Flux-ΔFlux-Flow trip setpoints from 4 hours to 10 hours (ITS 3.2.5, DOC L02). CTS 3.2.2 Action A states the High Flux and Flux-ΔFlux-Flow trip setpoints must be reduced 1 percent for each 1 percent Nuclear Heat Flux Hot Channel Factor exceeds its limit within 4 hours. CTS 3.2.3 Action A states the High Flux and Flux-ΔFlux-Flow trip setpoints must be reduced to 1 percent for each 1 percent Nuclear Enthalpy Rise Hot Channel Factor exceeds its limit within 4 hours. ITS 3.2.5 Required Actions A.2 and B.2 requires the trip setpoints to be reduced similarly within 10 hours.

G.2.4.1 Regulatory Evaluation

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of structures, systems and components that are required to protect the health and safety of the public. The Commission's regulatory requirements that are related to the content of the TSs are contained in 10 CFR 50.36, "Technical Specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) SLs, LSSSs and control settings; (2) LCO; (3) SR; (4) design features; and (5) administrative controls.

The NRC staff also applied the following regulatory requirement in reviewing the application:

GDC 15, as it relates to the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

G.2.4.2 Technical Evaluation

The licensee proposes to increase the CT to reduce the High Flux and Flux-ΔFlux-Flow trip setpoints when either power peaking factors (F_Q or $F^N_{\Delta H}$) are outside of its limits. The power peaking factors establish limits that constrain the core power distribution within design limits during normal operation and during anticipated operational occurrences such that accident initial condition protection criteria are preserved. The accident initial condition criteria are preserved by bounding operation at thermal power within specified acceptable fuel design limits. The F_Q limit is a specified acceptable fuel design limit that preserves the initial conditions for the ECCS analysis. The $F^N_{\Delta H}$ limit is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss of flow transient.

The CTS states the High Flux and Flux-ΔFlux-Flow trip setpoints must be reduced 1 percent for each 1 percent F_Q exceeds its limit within 4 hours. Also, the CTS states that the High Flux and Flux-ΔFlux-Flow trip setpoints must be reduced 1 percent for each 1 percent $F^N_{\Delta H}$ exceeds its limit within 4 hours. ITS 3.2.5 Required Actions A.2 and B.2 requires the trip setpoints to be reduced similarly within 10 hours. This proposed change is similar to BSI-4, ITS 3.4.4, RCS Loops - MODES 1 and 2 in which it was requested to increase the CT to 10 hours for reducing the High Flux trip setpoint.

Based on the similar proposed changes in BSI-4, ITS 3.4.4 and the NRC staff's evaluation, an increase of the CT to 10 hours is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the allowed CT. The NRC staff also evaluated ITS Table 3.3.1-1, "Reactor Protection System Instrumentation" Functions to ensure that they provided the same level of protection as the STS Table 3.3.1-1, "Reactor Protection System Instrumentation" Functions. The NRC staff found the two tables to be consistent and provide the same level of protection. Therefore, the NRC staff finds the proposed changes acceptable.

G.2.4.3 Conclusion

The NRC staff has reviewed BSI-9 related to TS changes in the DBNPS ITS Conversion. Based on its review of the licensee's submittal and responses to the RAIs, the NRC staff finds that the proposed TS changes related to BSI-9 are acceptable.

G.2.5 BSI-13: ITS 3.3

BSI-13 proposes the following changes related to draft TSTF-493:

a. Adds Footnotes (c) and (d) to ITS Table 3.3.1-1 Functional Unit 1a (ITS 3.3.1, Attachment 1 Volume 8, page 43 of 636 of application).

b. Allows Method 1 or Method 2 of ISA 67.04-Part II - 1994 or ISA 67.04.02 - 2000 for all RPS Functional Units in the ITS Bases (ITS 3.3.1 Attachment 1 Volume 8, page 59 of 636 of application).

c. Allows modification to where the Nominal trip setpoints are specified in the TS Bases (ITS 3.3.1 Attachment 1 Volume 8, pages 60 and 62 of 636 of application).

d. Adds a statement concerning setpoint methodology to the Bases in the ITS (ITS 3.3.1 Attachment 1 Volume 8, pages 81-84 of 636 of application).

e. Allows Method 1 or Method 2 of ISA 67.04-Part II - 1994 or ISA 67.04.02 - 2000 for all Safety Features Actuation System (SFAS) Functional Units in the ITS Bases (ITS 3.3.5 Attachment 1 Volume 8, page 209 of 636 of application).

f. Allows Method 1 or Method 2 of ISA 67.04-Part II - 1994 or ISA 67.04.02 - 2000 for all SFRCS Functional Units in the ITS Bases (ITS 3.3.11 Attachment 1 Volume 8, pages 394-395 of 636 of application).

G.2.5.1 Regulatory Evaluation

The regulatory requirements and guidance which the NRC staff considered in its review of the application are as follows:

Part 50 of 10 CFR establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

GDC - 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC - 13, "Instrumentation and control," requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

GDC - 20, "Protective system functions," requires the protection system be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Section 50.36 of 10 CFR - "Technical Specifications," states, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section."

> Specifically, 10 CFR 50.36(c)(1)(ii)(A) requires, in part, where a limited safety system setting (LSSS) is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.
>
> Additionally, 10 CFR 50.36(c)(3) requires, surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that LCOs will be met.

RG 1.105, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

RIS 2006-17, "A NRC Staff Position on the Requirements of 10 CFR 50.36 regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006, provides additional clarification on the requirements of 10 CFR 50.36.

G.2.5.2 Technical Evaluation

The licensee proposed the inclusion of information in the ITS TS and Bases that was not included in the CTS or the ISTS. The issues related to this information have been classified as out of scope issues (OSIs). For each OSI, the NRC staff asked a question and the licensee provided a response. The questions and responses have been included in the licensee's letter dated August 26, 2008 (ADAMS Accession No. ML082600594). The RAIs are organized by ITS Section and then by the thread number (see number listed in parentheses below). EICB reviewed the licensee responses for OSIs 22, 23, 25, 29, 46, and 64.

G.2.5.2.1 OSI 22 (20071160953)

Footnotes (c) and (d) apply to ITS SR 3.3.1.5 and SR 3.3.1.7 for ITS RPS Table 3.3.1-1 Function 5, Reactor Coolant (RC) Pressure-Temperature [CTS Table 4.3-1 Note 10 for CTS Function 7, RC Pressure-Temperature]. The licensee proposed that Footnotes (c) and (d) also apply to ITS SR 3.3.1.3 for ITS Table 3.3.1-1, Function 1.a, High Flux High Setpoint [CTS Table 4.3-1 Function 2, High Flux].

The purpose of Footnotes (c) and (d) is to follow RIS 2006-17, for LSSSs that protect the safety limit. The footnotes provide measures to be taken to assess the operability of LSSS instrumentation that protect the safety limit. These footnotes are only being applied to LSSSs that protect a safety limit and are being revised. The NRC staff finds that the application of Footnotes (c) and (d) to ITS SR 3.3.1.3 for ITS Table 3.3.1-1, Function 1.a, High Flux High Setpoint is in accordance with 10 CFR 50.36 and is, therefore, acceptable.

G.2.5.2.2 OSI 23 (200711160956)

The licensee proposed that ITS Bases B 3.3.1 describing Trip Setpoint/AVs include the statement, "The trip Setpoint is established using Method 1 or Method 2 of Reference 6 [ISA 67.04-1994] or Reference 7 [ISA 67.04-2000.]" CTS Bases 3/4.3.1 and 3/4.3.2 state that except for CTS RPS Table 4.3-1, Function 7, RC Pressure-Temperature, "Only the Allowable Value is specified for each Function," without providing details about the methodology used to determine the AV. For CTS Table 4.3.1 Function 7, the Bases for CTS 3/4.3.1 and 3/4.3.2 state, "The Limiting Trip Setpoint is specified in the UFSAR Technical Requirements Manual and the Limiting Trip Setpoint may be established using Method 1 or Method 2 ..."

ISA 67.04-1994 is endorsed by RG 1.105. ITS Bases B 3.3.1 provides the methodology used for all ITS 3.3.1 setpoints and AVs. This information is more explicit than the information in the CTS Bases. Therefore, the NRC staff concludes that the inclusion of, "The trip Setpoint is established using Method 1 or Method 2 of Reference 6 [ISA 67.04-1994] or Reference 7 [ISA 67.04-2000.]," in ITS Bases B 3.3.1 is acceptable.

G.2.5.2.3 OSI 25 (200711160940)

The licensee proposed that ITS Bases B 3.3.1 include additional information concerning ITS RPS Table 3.3.1-1 Function 1.a, High Flux High Setpoint and Function 5, RC Pressure - Temperature, related to Footnotes (c) and (d). For these two functions, ITS Bases B 3.3.1 includes information that the Limiting Trip Setpoint, the methodology used to determine the Limiting Trip Setpoint, the pre-defined as-found acceptance criteria, and the as-left tolerance, are specified in the Technical Requirements Manual.

The ITS Bases B 3.3.1 statements on pages B 3.3.1-11 and B 3.3.1-12 provide additional information concerning ITS Table 3.3.1-1, Function 1.a and Function 5, related to Footnotes (c) and (d). For these two functions, the inclusion, in ITS Bases B 3.3.1, of information that the Limiting Trip Setpoint, the methodology used to determine the Limiting Trip Setpoint, the pre-defined as-found acceptance criteria, and the as-left tolerance, are specified in the Technical Requirements Manual, follows the recommendations of RIS 2006-17 and therefore the NRC staff finds these changes acceptable.

G.2.5.2.3 OSI 29 (200711161018)

The licensee proposed adding a description of how ITS RPS SR 3.3.1.5 and SR 3.3.1.7 Footnotes (c) and (d) are applied for ITS Table 3.3.1-1 Function 5, RC Pressure-Temperature. The NRC staff finds that although this detailed information is not included in CTS Bases 3/4.3.1 and 3/4.3.2, this information provides greater detail than CTS Bases 3/4.3.1 and 3/4.3.2, is consistent with the goal of RIS 2006-17, and is, therefore, acceptable.

G.2.5.2.4 OSI 46 (200711161110 and 200711160956)

For ITS SFAS Table 3.3.5-1 functions, ITS Bases B 3.3.5 states, "The trip setpoint is established using Method 1 or Method 2 ..." CTS Bases 3/4.3.1 and 3/4.3.2 state that for CTS

SFAS Table 3.3-4, "Only the Allowable Value is specified for each Function," without providing details about the methodology used to determine the Allowable Values. The NRC staff concludes that the information in ITS Bases B 3.3.5 is more explicit than the information in CTS Bases 3/4.3.1 and 3/4.3.2, and is, therefore, acceptable.

G.2.5.2.5 OSI 64 (200801101044)

For the ITS Steam SFRCS Table 3.3.11-1 functions, ITS Bases B 3.3.11 states, "The trip setpoint is established using Method 1 or Method 2 ..." CTS Bases 3/4.3.1 and 3/4.3.2 states that for CTS SFRCS Table 3.3-12, Function 2, Steam Generator Level-Low, "Only the Allowable Value is specified for each Function," without providing details about the methodology used to determine the AV. The NRC staff finds that the information in ITS Bases B 3.1.11 is more explicit than the information in CTS Bases 3/4.3.1 and 3/4.3.2, and is, therefore, acceptable.

G.2.5.3 Conclusion

The NRC staff has reviewed the above stated OSIs related to TS changes in the ITS Conversion of the DBNPS. Based on its review of the licensee's submittal and responses to the RAIs, the NRC staff finds that the proposed TS changes related to the above stated OSIs are acceptable.

G.2.6 BSI-19: ITS 3.3.15

BSI-19 proposes to delete CTS 3/4.9.4, "Containment Penetrations, " concerning the Containment Purge and Exhaust Isolation TSs.

G.2.6.1 Regulatory Evaluation

Section 50.36(c)(2)(i) of 10 CFR states "limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility."

Section 50.36(c)(2)(ii) of 10 CFR further states "a technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

> Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

G.2.6.2 Technical Evaluation

One function of the containment is to minimize the release of fission product radioactivity to the environment as a result of fuel element rupture. CTS LCO 3/4.9.4, "Containment Penetrations" is applicable during core alterations or movement of irradiated fuel within the containment. The only accident postulated to occur during core alterations that results in a significant radioactive release is the fuel handling accident. However, containment isolation is not assumed in the fuel handling accident inside containment as documented in UFSAR Section 15.4.7 and Table 15.4.7-4a. Therefore the NRC staff finds that the deletion of CTS LCO 3/4.9.4 from the TS is acceptable.

G.2.6.3 Conclusion

The NRC staff reviewed BSI-19 related to a TS change in the ITS conversion of the DBNPS. Based on its review of the licensee's submittal and response to the NRC staff's questions, the NRC staff finds that the proposed TS change related to BSI-19 is acceptable.

G.2.7 BSI-21: ITS 3.3.15 DOC M03

BSI-21 proposes to deviate from the STS by not placing the Control Room Emergency Ventilation System (CREVS) in operation during the movement of irradiated fuel for an inoperable channel, and not immediately suspending irradiated fuel movements if two channels are inoperable and compensatory actions are not immediately carried out (ITS 3.3.15 DOC M03).

G.2.7.1 Regulatory Evaluation

This SE input discusses the impact of the proposed changes on the previously analyzed radiological consequences of design-basis accidents. The regulatory requirements against which the Accident Dose Branch (AADB) performed its review of the licensee's current request are the accident dose criteria in 10 CFR 100.11 and 10 CFR Part 50 Appendix A, GDC 19, "Control room." The AADB staff also considered the relevant information in the DBNPS UFSAR.

The regulatory requirements and guidance which the Containment and Ventilation Branch staff considered in its review of the application are as follows:

Part 50 of 10 CFR establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

Paragraph 50.36(c)(2)(ii) of 10 CFR, "Technical Specifications," requires that a TS LCO of a nuclear reactor must be established for each item meeting one or more of the criteria set forth in 10 CFR 50.36(c)(2)(ii)(A)-(D).

Paragraph 50.36(c)(3) of 10 CFR, "Technical Specifications," requires that TSs include SRs, which "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

Section 50.59 of 10 CFR, "Changes, Tests, and Experiments".

Appendix "A" of Part 50 of 10 CFR, "General Design Criteria for Nuclear Power Plants",

GDC 19, "Control Room", provides for a control room from which actions can be taken to maintain the nuclear power plant in a safe condition under accident conditions.

GDC 60, "Control of Releases of Radioactive Materials to the Environment", requires the means to control the release of radioactive materials in gaseous and liquid effluents.

GDC 64, "Monitoring Radioactivity Releases", requires means for monitoring effluent discharge paths for radioactivity that may be released from normal operations, including AOOs, and from postulated accidents.

Part 100.11 of 10 CFR, "Determination of the exclusion area, low population zone, and population center distance." This regulation provides requirements for the protection of an individual located on the plant's boundary for two hours immediately following onset of the postulated fission product release.

NUREG-0800, Standard Review Plan Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents"

NUREG-1430, Rev. 3.0, "Standard Technical Specifications Babcock and Wilcox Plants"
RG 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments"

RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors"

G.2.7.2 Technical Evaluation

UFSAR Section 15.4.7 describes that the control room (CR) is assumed to be isolated during a fuel handling accident (FHA). The FHA radiological consequences analysis, both inside and outside of containment, assumes the Control Room Normal Ventilation System is isolated by the Station Vent Normal Range Radiation Monitoring high radiation signal. The purpose of ITS 3.3.16 is to provide assurance that the Station Vent Normal Radiation Monitoring Instrumentation is operable when required to perform its function.

ISTS LCO 3.3.16 includes a requirement to have one channel of Control Room Isolation-High Radiation operable. However, the number of channels "One" is bracketed. CTS 3/4.7.6.1 requires two channels to be operable, therefore, the licensee changed the required number of channels in ITS LCO 3.3.16 from "One" to "Two" channels. The ISTS 3.3.16 Actions only include an action (Action A) for one channel inoperable. As a result, the licensee modified the Actions to reflect the CLB (CTS 3.7.6.1 Actions B and C). The licensee stated that the requirement to "Isolate the Control Room Normal Ventilation System" has been added as Required Action A.1 in order to be consistent with the current requirements.

The CT of Required Action A.1 has been extended from 1 hour to 7 days and a new Action (ITS 3.3.15 Action B) has been added. The licensee states that because the CREVS is not required to be operable during movement of irradiated fuel assemblies, it added Required Actions A.2 and B.2 to only require CREVS to be placed in operation in MODES 1, 2, 3, and 4 which is consistent with the requirements for the CREVS. The ISTS Required Action D.1 has not been included for this same reason.

The NRC staff finds that the licensee's proposed changes, as described above, are consistent with the CLB. The current DBNPS FHA radiological consequence analysis remains unaffected by the proposed changes. Therefore, the NRC staff concludes that these changes are acceptable with respect to the radiological consequences of design-basis accidents.

The licensee proposed in TS 3.3.15, "Station Vent Normal Range Radiation Monitoring", Required Action B.1, (for two channels of normal station vent radiation monitors inoperable), "isolate the control room normal ventilation system" with a CT of one hour. The licensee also proposed a new Required Action D.1, immediate suspension of irradiated fuel assembly movement if Condition A or B are not met.

The NRC staff requested clarification regarding the difference between the proposed TS and NUREG-1430, Rev. 3.0, (B&W Plants STS) model for ISTS 3.3.16 which does not permit any grace time to return at least one channel to operability but instead requires immediately placing one OPERABLE CREVS train in emergency operation mode or immediate suspension of the movement of irradiated fuel. This was a concern since the licensee is adopting the use of the term "recently irradiated" for fuel that has occupied part of a critical reactor core within the previous 72 hours. In addition, the licensee had proposed, in Rev. 0, to remove current TS 3.9.3, "Decay Time", which requires the reactor to be subcritical 72 hours before spent fuel movement of irradiated fuel in the reactor pressure vessel.

The licensee responded on 5/29/2008 (Note: The ITS number was changed from ITS 3.3.16 to ITS 3.3.15 in Rev. 1.):

> The current licensing basis at Davis-Besse, as shown in CTS 3.7.6.1 (Volume 8, Page 518) does not require the Station Vent Normal Range Radiation Monitoring to be OPERABLE during movement of irradiated fuel assemblies. Davis-Besse added this new Applicability to ITS 3.3.16 (Page 524) as justified in DOC M03 (Page 521). As part of this addition, ACTIONS for inoperable channels when moving irradiated fuel (i.e., ACTIONS A, B, and D) were also added. Thus, the addition of ITS 3.3.16 ACTIONS A, B, and D (during movement of irradiated fuel) is not a less restrictive change, but a more restrictive change. ITS 3.3.16 Required Action B.1 (Page 525) allows 1 hour to isolate the Control Room Normal Vent System. This Required Action applies during MODES 1, 2, 3, and 4, and also during movement of irradiated fuel. Davis-Besse believes that since 1 hour is provided in CTS 3.7.6.1 Action C for when both channels are inoperable in MODE 1, 2, 3, or 4, then the same 1 hour is acceptable when moving irradiated fuel assemblies. This 1 hour time was approved by the NRC as documented in the Safety Evaluation for Amendment 227, dated October 5, 1998. However, Davis-Besse has noted that DOC M03 does not clearly state that the addition of ACTIONS A and B, as they relate to moving irradiated fuel, is part of DOC M03. Therefore, DOC M03 will be revised to clearly describe the entire more restrictive change. A draft markup regarding this change is attached. This change will be reflected in the supplement to this section

of the ITS Conversion Amendment. The NRC reviewer also requested that Davis-Besse include in the discussion Control Room Habitability and the movement of fuel that has occupied part of a critical core within the previous 72 hours. The Davis-Besse accident analysis does not assume any irradiated fuel movement prior to 72 hours. Fuel movement prior to this time is currently precluded by CTS 3.9.3 (Volume 14, Page 128). Davis-Besse is relocating this current requirement to the Technical Requirements Manual (TRM), consistent with NUREG-1430. The NUREG does not include this Specification. The TRM is currently incorporated into the UFSAR, thus is controlled by the requirements of 10 CFR 50.59. Davis-Besse expects to receive a License Condition that all changes covered by LA type and R type Description of Change (DOC), which include CTS 3.9.3, be moved to the location specified in the applicable DOC (in this specific case, the TRM) and controlled by the process specified in the DOC (in this case, 10 CFR 50.59) as part of the ITS amendment approval.

In response to the NRC staff's RAI for a different issue, (RAI No. 200801161532) the licensee indicates that they will not be moving current TS 3.9.3 to the technical requirements manual (TRM). Instead of moving the requirement for delay time before moving irradiated fuel to the TRM the delay time for fuel movement will remain controlled by a technical specification. The licensee stated that the movement of fuel in the reactor vessel will not occur unless the reactor has been subcritical for greater than 72 hours.

The NRC staff finds the clarifications by the licensee to be acceptable. The one hour delay to isolate the Control Room Normal Ventilation System and one hour delay to place one OPERABLE CREVS train in operation is consistent with the existing licensing basis. The existing plant fuel handling analysis shows that a fuel handling accident involving fuel that has been in the sub-critical reactor vessel for greater than 72 hours will not cause the radiation exposure to occupants of the control room to exceed the limits of GDC 19. Offsite radiation exposure remains well within the limits of 10 CFR 100.11. Based on the above evaluation, the proposed change is acceptable.

G.2.7.3 Conclusion

As described above, the NRC staff reviewed the justifications used by the licensee to assess the radiological impacts of deviations from ITS 3.3.16 "Station Vent Normal Range Radiation Monitoring." The NRC staff finds that the licensee used methods consistent with the regulatory requirements and guidance identified in Section G.2.7.1 above. The NRC staff finds, with reasonable assurance that the licensee's estimates of the exclusion area boundary, low-population zone, and control room doses will continue to comply with these criteria. Therefore, the proposed TS changes are acceptable with regard to the radiological consequences of postulated design-basis accidents.

Based on the above evaluation the NRC staff finds the proposed changes to the DBNPS TSs provide reasonable assurance of the ability to mitigate the effects a postulated fuel handling accident. The NRC staff also concludes that the proposed TS changes are in accordance with 10 CFR 50.36, and the requirements of GDCs 19, and 10 CFR 100.11. Therefore, the NRC staff finds the proposed change acceptable.

G.2.8 BSI-22: ITS 3.3.8

BSI-22 proposes a new definition of LOPS operability in the TS Bases (ITS 3.3.8 Attachment 1 Volume 8, page 298 of 636 of application).

G.2.8.1 Regulatory Evaluation

The NRC staff considered the following regulatory requirements and guidance in its review of the application:

Part 50 of 10 CFR, "Domestic Licensing of Production and Utilization Facilities," establishes the fundamental regulatory requirements. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

GDC 10, "Reactor Design," in Appendix A to 10 CFR Part 50 requires that "the reactor core and associated coolant, control, and protection systems ... be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

GDC 13, "Instrumentation and control," in Appendix A to 10 CFR Part 50 requires that "instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, [AOOs], and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."

GDC 21, "Protection system reliability and testability," in Appendix A to 10 CFR Part 50 requires that "the protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated."

Section 50.36 of 10 CFR, "Technical Specifications," states, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section."

G.2.8.2 Technical Evaluation

The STS for LCO 3.3.8 of NUREG-1430 is based on a design that utilizes a two-out-of-three logic design. The licensee's current logic design utilizes a one-out-two taken twice logic design for both loss of voltage and degraded voltage relays. The licensee proposes to change current TS 3/4.3.2 Safety System Instrumentation to reflect their current logic design for the loss of voltage and degraded voltage instrumentation for the EDG LOPS function. The NRC has classified the issue related to this information as BSI-22. The NRC staff asked a question related to single failure criteria for the system and the licensee provided the response. They both appear on the NRC/Davis-Besse ITS Conversion website (200712260942).

The licensee's current design has four undervoltage relays per bus arranged in a one-out-of-two taken twice logic. Each one of the one-out-of-two logic relays energizes an auxiliary relay. For the diesel start, load sequencer, and load shed, both auxiliary relays have to actuate. Loss of either relay could prevent this function. However, this will result in a loss of one DG, but the other DG will remain operable. Therefore, this meets the single failure requirements of GDC 21. This design has been reviewed and approved previously by the NRC staff. Therefore, the NRC staff finds the proposed change acceptable.

G.2.8.3 Conclusion

The NRC staff reviewed BSI-22 related to a TS Bases change in the ITS Conversion of the DBNPS. Based on its review of the licensee's submittal and responses to the NRC staff's questions, the NRC staff finds that the proposed TS Bases change related to BSI-22 is acceptable.

5.0 DELETED LICENSE CONDITIONS

License Condition 2.C(5), the secondary water chemistry monitoring program, is proposed to be deleted. This is acceptable since the requirements of this License Condition have been included in ITS 5.5.9, "Secondary Water Chemistry Program."

6.0 LICENSEE COMMITMENTS

In reviewing the proposed ITS Conversion for DBNPS, the NRC staff has relied upon the licensee's commitment to relocate certain requirements from the CTS to licensee-controlled documents as described in Table LA, "Removed Details" (Attachment 4 to this SE) and Table R, "Relocated Specifications" (Attachment 6 to this SE). These tables, and Sections 4.D and 4.E of this SE, reflect the relocations described in the licensee's submittals on the conversion. The NRC staff requested and the licensee submitted a set of license conditions to make these commitments enforceable (see Section 7.0 of this SE). Such commitments from the licensee are important to the ITS Conversion because the acceptability of removing certain requirements from the TSs is based on those requirements being relocated to licensee-controlled documents where further changes to the requirements will be controlled by applicable regulations or other requirements (e.g., 10 CFR 50.59).

7.0 LICENSE CONDITIONS

In its letter dated August 7, 2008, the licensee agreed to license conditions which describe (1) the relocation of certain CTS requirements and license conditions, as applicable, to other license controlled documents prior to ITS implementation, and (2) a schedule to begin performing new and revised SRs after ITS implementation. The following license conditions are included in the Facility Operating Licenses (FOLs):

1. This amendment authorizes the relocation of certain TS requirements and operating license conditions to other licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the other documents,

as described in (1) Sections D and E of the NRC staff's Safety Evaluation, and (2) Table LA of Removed Details and Table R of Relocated Specifications attached to the NRC staff's Safety Evaluation, which is enclosed with this amendment.

2. The schedule for performing the new or revised SRs in License Amendment No. 279 shall be as follows:

   For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment.

   For SRs that existed prior to this amendment whose intervals of performance are being reduced the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

   For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

   For SRs that existed prior to this amendment whose intervals of performance are being extended the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.

The NRC staff has reviewed the above schedule for the licensee to begin performing the new and revised SRs and concludes that it is acceptable. The licensee states that its implementation date for the new ITSs will be no later than 180 days from the date of issuance. This implementation date is acceptable.

Because the commitments discussed in Section 6.0 of this SE are being relied upon for the amendment, a license condition is included in the amendment that will enforce the relocation of requirements from the CTSs to licensee-controlled documents. The relocations are described in Table LA and Table R, which are Attachments 4 and 6 to this SE. The license condition states that implementation of this amendment shall include relocation of these requirements to the specified documents. The relocation of these requirements to the specified documents is to be completed upon implementation of the ITS. This implementation date is acceptable.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

9.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the *Federal Register* on October 20, 2008 (73 FR 62343), for the proposed conversion of the CTS to ITS for DBNPS. Accordingly, the Commission has determined that issuance of these amendments will not result in any significant environmental impacts other than those evaluated in the Final Environmental Statement for DBNPS dated

October 1975. The Commission also issued a Notice of Consideration of Issuance of Amendment to FOLs and Opportunity for a Hearing on May 22, 2008 (73 FR 29787-29791). There have been no comments or requests for hearing.

10.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachments:
1. List of Standard Acronyms and Abbreviations
2. Table A - Administrative Changes
3. Table L - Less Restrictive Changes
4. Table LA - Removed Details
5. Table M - More Restrictive Changes
6. Table R - Relocated Specifications

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Date: November 20, 2008

November 20, 2008

A copy of the related SE is enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions concerning this letter and the SE, contact me at 301-415-3719 or email Cameron.Goodwin@nrc.gov.

Sincerely,

/RA/

Cameron S. Goodwin, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 279 to NPF-3
2. Safety Evaluation

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Package Accession Number: ML082900616
Amendment Accession Number: ML082900600
List of Acronyms Accession Number: ML082900630
Table A Accession Number: ML082900683
Table L Accession Number: ML082900780
Table LA Accession Number: ML082900882
Table M Accession Number: ML082910112
Table R Accession Number: ML08 2910125

Note: Safety evaluation input on beyond-scope issues has been provided by the respective technical branches. The memos transmitting the SEs may be found in ADAMS under the TAC Nos. listed in the title above.

OFFICE	LPL3-2/PM	LPL3-2/LA	Tech Branch	OGC	DIRS/ITSB	LPL3-2/BC
NAME	CGoodwin	EWhitt	See Note	JBielecki	RElliott	RGibbs
DATE	11/19/08	11/19/08	Various	11/19/08	10/30/08	11/20/08

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