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

July 24, 2025

Ms. Jean A. Fleming
Vice President, Licensing,
Regulatory Affairs, and PSA
Holtec International, LLC
Krishna P. Singh Technology Campus
1 Holtec Boulevard
Camden, NJ 08104

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT NO. 276 RE: AMENDMENT REQUEST TO REVISE RENEWED FACILITY OPERATING LICENSE AND PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS TO SUPPORT RESUMPTION OF POWER OPERATIONS (EPID L-2023-LLA-0174)

Dear Ms. Fleming:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 276 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant (PNP). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 14, 2023, as supplemented by letters dated July 9, 2024, and December 19, 2024. The amendment revises the renewed facility operating license, the permanently defueled TS, the environmental protection plan, and the physical security plan to reflect the reauthorization of power operations at PNP.

The NRC staff has separately reviewed and approved Holtec's license transfer application, exemption request, and three license amendment requests related to the resumption of power operations at PNP. The NRC staff is issuing its approval of these actions concurrently with its approval of this amendment to reauthorize power operations at PNP.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Justin C. Poole, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 276 to DPR-20
2. Safety Evaluation

cc: Listserv

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

HOLTEC PALISADES, LLC

PALISADES ENERGY, LLC

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 276
License No. DPR-20

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

   A. The application for amendment filed by Holtec Decommissioning International, LLC,[1] on behalf of Holtec Palisades, LLC, dated December 14, 2023, as supplemented by letters dated July 9, 2024, and December 19, 2024, 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 set forth in 10 CFR Chapter I;

   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, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the Commission's regulations and all applicable requirements have been satisfied.

[1] By letter dated July 24, 2025, the NRC issued Amendment No. 275, reflecting Palisades Energy, LLC, as the licensed operator (the licensee) for Palisades Nuclear Plant.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-20 is hereby amended to read as follows:

   (2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 276, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Palisades Energy shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective upon the licensee's submittal of a request to rescind the 10 CFR 50.82(a)(1) certifications and shall be implemented within 30 days from the amendment effective date.

FOR THE NUCLEAR REGULATORY COMMISSION

ILKA BERRIOS

Digitally signed by ILKA BERRIOS
Date: 2025.07.24 12:46:36 -04'00'

Ilka Berrios, Acting Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: July 24, 2025

ATTACHMENT TO LICENSE AMENDMENT NO. 276

PALISADES NUCLEAR PLANT

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Renewed Facility Operating License No. DPR-20

Replace the following pages of Renewed Facility Operating License No. DPR-20 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating areas of change.

REMOVE	INSERT
Pages 1 through 7	Pages 1 through 8

Technical Specifications

Replace the following pages of the Appendix A Permanently Defueled 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
TS Title Page	TS Title Page
Page i	None
Pages 1.1-1	Page 1.1-1 through 1.1-7
Pages 1.2-1 through 1.2-2	Pages 1.2-1 through 1.2-3
Pages 1.3-1 through 1.3-2	Pages 1.3-1 through 1.3-12
Pages 1.4-1 through 1.4-3	Pages 1.4-1 through 1.4-8
Page 2.0-1	Page 2.0-1
Pages 3.0-1 through 3.0-2	Pages 3.0-1 through 3.0-6
-------------------	Pages 3.1.1-1 through 3.7.13-1
Page 3.7.14-1	Page 3.7.14-1
Page 3.7.15-1	Page 3.7.15-1
Page 3.7.16-1	Page 3.7.16-1
-------------------	Pages 3.7.17-1 through 3.9.6-1
Pages 4.0-1 through 4.0-5	Pages 4.0-1 through 4.0-6
Pages 5.0-7 through 5.0-15	Pages 5.0-7 through 5.0-30

Environmental Protection Plan

Replace the following pages of the Appendix B Environmental Protection Plan with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Cover Sheet	Cover Sheet
Table of Contents	Table of Contents
Page 1-1	Page 1-1
Page 2-1	Page 2-1
Pages 3-1 through 3-3	Pages 3-1 through 3-3
Page 4-1	Page 4-1
Pages 5-1 through 5-4	Pages 5-1 through 5-4

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HOLTEC PALISADES, LLC

PALISADES ENERGY, LLC

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-20

1. The Nuclear Regulatory Commission (NRC or the Commission) having previously made the findings set forth in Operating License No. DPR-20, dated February 21, 1991, has now found that:

   A. The application for Renewed Operating License No. DPR-20 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, and all required notifications to other agencies or bodies have been duly made;

   B. [deleted];

   C. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations;

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

E. There is reasonable assurance: (i) that the activities authorized by this renewed operating license 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 set forth in 10 CFR Chapter I;

F. Holtec Palisades, LLC (Holtec Palisades) is financially qualified and Palisades Energy, LLC (Palisades Energy) is financially and technically qualified to engage in the activities authorized by this renewed operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

G. Holtec Palisades and Palisades Energy have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations;

H. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this renewed Facility Operating License No. DPR-20, subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 (formerly Appendix D to Part 50), of the Commission's regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this renewed operating license will be in accordance with 10 CFR Parts 30, 40, and 70.

2. Renewed Facility Operating License No. DPR-20 is hereby issued to Holtec Palisades and Palisades Energy as follows:

   A. This renewed license applies to the Palisades Plant, a pressurized light water moderated and cooled reactor and electrical generating equipment (the facility). The facility is located in Van Buren County, Michigan, and is described in the Palisades Plant Updated Final Safety Analysis Report, as supplemented and amended, and in the Palisades Plant Environmental Report, as supplemented and amended.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

      (1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec Palisades to possess and use, and (b) Palisades Energy, LLC (Palisades Energy) to possess, use and operate the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) Palisades Energy, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) Palisades Energy, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) Palisades Energy, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) Palisades Energy, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; 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) Palisades Energy is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 276, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Palisades Energy shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection
Palisades Energy shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment requests dated December 12, 2012, November 1, 2017, November 1, 2018, and March 8, 2019, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014, August 14, 2014, November 4, 2014, December 18, 2014, January 24, 2018, and May 28, 2019, as approved in the safety evaluations dated February 27, 2015, February 27, 2018, and August 20, 2019. Except where NRC approval for changes or

deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated February 27, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as

specified by 2, below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-2, "Plant Modifications Committed," of Entergy Nuclear Operations, Inc. (ENO) letter PNP 2019-028 dated May 28, 2019, to complete the transition to full compliance with 10 CFR 50.48(c) before the end of the refueling outage following the fourth full operating cycle after NRC approval. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Table S-3, "Implementation Items," of ENO letter PNP 2014-097 dated November 4, 2014, within six months after NRC approval, or six months after a refueling outage if in progress at the time of approval with the exception of Implementation Items 3 and 8 which will be completed once the related modifications are installed and validated in the PRA model.

(4) [deleted]

(5) [deleted]

(6) Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

a. Fire fighting response strategy with the following elements:
   1. Pre-defined coordinated fire response strategy and guidance
   2. Assessment of mutual aid fire fighting assets
   3. Designated staging areas for equipment and materials
   4. Command and control
   5. Training of response personnel

b. Operations to mitigate fuel damage considering the following:
   1. Protection and use of personnel assets
   2. Communications
   3. Minimizing fire spread
   4. Procedures for implementing integrated fire response strategy
   5. Identification of readily-available pre-staged equipment
   6. Training on integrated fire response strategy
   7. Spent fuel pool mitigation measures

c. Actions to minimize release to include consideration of:
   1. Water spray scrubbing
   2. Dose to onsite responders

(7) [deleted]

(8) Amendment 257 authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61.

D. The facility has been granted certain exemptions from Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted in a letter dated December 6, 1989.

These exemptions granted pursuant to 10 CFR 50.12, are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Palisades Energy shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Palisades Nuclear Plant Physical Security Plan."

Palisades Energy shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Palisades CSP was approved by License Amendment No. 243 as supplemented by changes approved by License Amendment Nos. 248, 253, 259, and 264.

F. [deleted]

G. Holtec Palisades and Palisades Energy shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. The Updated Safety Analysis Report supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the Updated Safety Analysis Report required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, Palisades Energy may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Palisades Energy evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

I. The Updated Safety Analysis Report supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. Palisades Energy shall complete these activities no later than March 24, 2011, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

J. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal scheduled, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

K. This license is effective as of the date of issuance and shall expire at midnight March 24, 2031.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A –Technical Specifications

2. Appendix B – Environmental Protection Plan

Date of Issuance: January 17, 2007

PALISADES PLANT

RENEWED FACILITY OPERATING LICENSE DPR-20

APPENDIX A

TECHNICAL SPECIFICATIONS

As Amended through Amendment No. 276

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.
AVERAGE DISINTEGRATION ENERGY - $\bar{E}$	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the primary 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.
AXIAL OFFSET (AO)	AO shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the incore monitoring system).
AXIAL SHAPE INDEX (ASI)	ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core (determined using the excore monitoring system).
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training and retraining program required by Specification 5.3.2.

1.1 Definitions

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.

Whenever a RTD or thermocouple sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

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 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:

a. Analog and bistable channels - 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;

b. Digital channels - the use of diagnostic programs to test digital hardware and the injection of simulated process data into the channel to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY

1.1 Definitions

CHANNEL FUNCTIONAL TEST (continued)	The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or control rods within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. 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 dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989; (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

1.1 Definitions

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal water leakoff), 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 not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; and

3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a PCS component body, pipe wall, or vessel wall.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average primary coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1.

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 Chapter 13, Initial Tests and Operation, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT (T_q)	T_q shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the primary coolant of 2565.4 MWt.
REFUELING BORON CONCENTRATION	REFUELING BORON CONCENTRATION shall be a Primary Coolant System boron concentration of ≥ 1720 ppm and sufficient to assure the reactor is subcritical by ≥ 5% $\Delta\rho$ with all control rods withdrawn.

1.1 Definitions

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 (shutdown and regulating) are fully inserted except for the single rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all full length control rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any full length control rods not capable of being fully inserted, the reactivity worth of these rods must be accounted for in the determination of SDM; and b. There is no change in part length rod position
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the primary coolant.
TOTAL RADIAL PEAKING FACTOR (F_R^T)	F_R^T shall be the maximum ratio of the individual fuel pin power to the core average pin power integrated over the total core height, including tilt.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[b]	< 0.99	NA	$300 > T_{ave} > 200$
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 AND and OR. 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 indentions 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.

1.2 Logical Connectors

EXAMPLES

The following examples illustrate the use of logical connectors.

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 plant. 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 discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the plant is in a MODE or specified condition stated in the Applicability of the LCO.

Unless otherwise specified, the Completion Time begins when a senior licensed operator on the operating shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. The "otherwise specified" exceptions are varied, such as a Required Action Note or Surveillance Requirement Note that provides an alternative time to perform specific tasks, such as testing, without starting the Completion Time. While utilizing the Note, should a Condition be applicable for any reason not addressed by the Note, the Completion Time begins. Should the time allowance in the Note be exceeded, the Completion Time begins at that point. The exceptions may also be incorporated into the Completion Time. For example, LCO 3.8.1, "AC Sources - Operating," Required Action B.2, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied.

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 plant 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

DESCRIPTION (continued)

associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.

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 not 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, unless otherwise specified.

However, when a subsequent 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 first inoperability; and

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

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 . . ." Example 1.3-3 illustrates one use of

DESCRIPTION (continued)

this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

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.

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.

1.3 Completion Times

EXAMPLES (continued)

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, Condition 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.

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 not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

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.

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

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 AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
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.	12 hours 12 hours

EXAMPLES

EXAMPLE 1.3-3 (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).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be 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. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

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 30 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 (including 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.

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

1.3 Completion Times

EXAMPLE (continued)

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.

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) 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.
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 performed 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” and “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.

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1.4 Frequency

DESCRIPTION (continued)

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:

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

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 plant is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility 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 plant 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 plant 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." The interval continues, whether or not the plant operation is < 25% RTP between performances. Therefore, if the Surveillance were 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 with power ≥ 25% RTP.

Once the plant reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, 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 plant 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 were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the plant 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 were not met), SR 3.0.4 would require satisfying the SR.

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 plant 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 were 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 plant 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 were 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 plant 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 were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the plant 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 were 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 Departure from Nucleate Boiling Ratio (DNBR) shall be maintained at or above the following DNB correlation safety limits:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

2.1.1.2 In MODES 1 and 2, the peak Linear Heat Rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kW/ft.

2.1.2 Primary Coolant System (PCS) Pressure SL

In MODES 1, 2, 3, 4, 5, and 6, the PCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, 5, or 6, restore compliance within 5 minutes.

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, LCO 3.0.8, and LCO 3.0.9.

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 plant 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 plant, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 31 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;

LCO 3.0.4 (continued)

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

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

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

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 5.5.13, "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 Special Test Exception (STE) LCOs in each applicable LCO section allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

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.

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

LCO 3.0.9 (continued)

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the 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. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. 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.

3.0 SR APPLICABILITY

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.

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 plant.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

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

APPLICABILITY: MODE 3, 4, and 5.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM to be within limits.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Balance

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

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 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.	7 days
	<u>AND</u>	
	A.2 Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.2.1	-----------------------------NOTE--------------------------- The predicted reactivity values may 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. --- Verify overall core reactivity balance is within ± 1% $\Delta\rho$ of predicted values.	Prior to entering MODE 1 after each fuel loading AND ----------NOTE--------- Only required after initial 60 EFPD -------------------------- In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained less positive than 0.5 E-4 Δρ/°F at ≤ 2% RATED THERMAL POWER (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 less positive than 0.5 E-4 Δρ/°F at ≤ 2% RTP.	Prior to exceeding 2% RTP after each fuel loading

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Alignment

LCO 3.1.4 All control rods, including their position indication channels, shall be OPERABLE and aligned to within 8 inches of all other rods in their respective group, and the control rod position deviation alarm shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel of rod position indication inoperable for one or more control rods.	A.1 Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes following any rod motion in that group
B. Rod position deviation alarm inoperable.	B.1 Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes of movement of any control rod
C. One control rod misaligned by > 8 inches.	C.1 Perform SR 3.2.2.1 (peaking factor verification). <u>OR</u> C.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours 2 hours
D. One full-length control rod immovable, but trippable.	D.1 Restore control rod to OPERABLE status.	Prior to entering MODE 2 following next MODE 3 entry

ACTIONS (continued)

E. Required Action and associated Completion Time not met. <u>OR</u> One or more control rods inoperable for reasons other than Condition D. <u>OR</u> Two or more control rods misaligned by > 8 inches. <u>OR</u> Both rod position indication channels inoperable for one or more control rods.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify the position of each control rod to be within 8 inches of all other control rods in its group.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.2	Perform a CHANNEL CHECK of the control rod position indication channels.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.3	Verify control rod freedom of movement by moving each individual full-length control rod that is not fully inserted into the reactor core ≥ 6 inches in either direction.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.4	Verify the rod position deviation alarm is OPERABLE.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.5	Perform a CHANNEL CALIBRATION of the control rod position indication channels.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.6	Verify each full-length control rod drop time is ≤ 2.5 seconds.	Prior to reactor criticality, after each reinstallation of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown and Part-Length Control Rod Group Insertion Limits

LCO 3.1.5 All shutdown and part-length rod groups shall be withdrawn to ≥ 128 inches.

APPLICABILITY: MODE 1,
MODE 2 with any regulating rod withdrawn above 5 inches.

---NOTE---
This LCO is not applicable while performing SR 3.1.4.3 (rod exercise test).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more shutdown or part-length rods not within limit.	A.1	Declare affected control rod(s) inoperable and enter the applicable Conditions and Required Actions of LCO 3.1.4.	Immediately
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown and part-length rod group is withdrawn ≥ 128 inches.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Regulating Rod Group Position Limits

LCO 3.1.6 The Power Dependent Insertion Limit (PDIL) alarm circuit and the Control Rod Out Of Sequence (CROOS) alarm circuit shall be OPERABLE, and the regulating rod groups shall be limited to the withdrawal sequence, overlap, and insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

---NOTE---
This LCO is not applicable while performing SR 3.1.4.3 (rod exercise test).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating rod groups inserted beyond the insertion limit.	A.1 Restore regulating rod groups to within limits. <u>OR</u> A.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the regulating rod group position and insertion limits specified in the COLR.	2 hours 2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Regulating rod groups not within sequence or overlap limits.	B.1 Restore regulating rod groups to within appropriate sequence and overlap limits.	2 hours
C. PDIL or CROOS alarm circuit inoperable.	C.1 Perform SR 3.1.6.1 (group position verification).	Once within 15 minutes following any rod motion
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify each regulating rod group is within its withdrawal sequence, overlap, and insertion limits.	In accordance with the Surveillance Frequency Control Program
SR 3.1.6.2	Verify PDIL alarm circuit is OPERABLE.	In accordance with the Surveillance Frequency Control Program
SR 3.1.6.3	Verify CROOS alarm circuit is OPERABLE.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Special Test Exceptions (STE)

LCO 3.1.7 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.4, "Control Rod Alignment";
LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits";
LCO 3.1.6, "Regulating Rod Group Position Limits"; and
LCO 3.4.2, "PCS Minimum Temperature for Criticality”

may be suspended, provided:

a. THERMAL POWER is ≤ 2% RTP;

b. ≥ 1% shutdown reactivity, based on predicted control rod worth, is available for trip insertion; and

c. T_{ave} is ≥ 500°F.

APPLICABILITY: MODE 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Reduce THERMAL POWER to within limit.	15 minutes
B. Shutdown reactivity not within limit.	B.1 Initiate boration to restore shutdown reactivity to within limit.	15 minutes
C. T_{ave} not within limit.	C.1 Restore T_{ave} to within limit.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Suspend PHYSICS TESTS.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify THERMAL POWER is ≤ 2% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.2	Verify T_{ave} is ≥ 500°F.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.3	Verify ≥ 1% shutdown reactivity is available for trip insertion.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Heat Rate (LHR)

LCO 3.2.1 LHR shall be within the limits specified in the COLR, and the Incore Alarm System or Excore Monitoring System shall be OPERABLE to monitor LHR.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR, as determined by the automatic Incore Alarm System, not within limits specified in the COLR, as indicated by four or more coincident incore channels. <u>OR</u> LHR, as determined by the Excore Monitoring System, not within limits specified in the COLR. <u>OR</u> LHR, as determined by manual incore detector readings, not within limits specified in the COLR.	A.1 Restore LHR to within limits.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Incore Alarm and Excore Monitoring Systems inoperable for monitoring LHR.	B.1	Reduce THERMAL POWER to ≤ 85% RTP.	2 hours
	AND		
	B.2	Verify LHR is within limits using manual incore readings.	4 hours AND Once per 2 hours thereafter
C. Required Action and associated Completion Time not met.	C.1	Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-------------------------------NOTE-------------------------- Only required to be met when the Incore Alarm System is being used to monitor LHR. --- Verify LHR is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	------------------------------NOTE---------------------------- Only required to be met when the Incore Alarm System is being used to monitor LHR. --- Adjust incore alarm setpoints based on a measured power distribution.	Prior to operation > 50% RTP after each fuel loading <u>AND</u> In accordance with the Surveillance Frequency Control Program
SR 3.2.1.3	------------------------------NOTE---------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. --- Verify measured ASI has been within 0.05 of target ASI for last 24 hours.	Prior to each initial use of Excore Monitoring System to monitor LHR
SR 3.2.1.4	------------------------------NOTE---------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. --- Verify THERMAL POWER is less than the APL.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.5	-------------------------------NOTE--------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. -- Verify measured ASI is within 0.05 of target ASI.	In accordance with the Surveillance Frequency Control Program
SR 3.2.1.6	-------------------------------NOTE--------------------------- Only required to be met when the Excore Monitoring System is being used to monitor LHR. -- Verify $T_q \leq 0.03$.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.2 TOTAL RADIAL PEAKING FACTOR (F_R^T)

LCO 3.2.2 F_R^T shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F_R^T not within limits specified in the COLR.	A.1 Restore F_R^T to within limits.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify F_R^T is within limits specified in the COLR.	Prior to operation > 50% RTP after each fuel loading <u>AND</u> In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.3 QUADRANT POWER TILT (T_q)

LCO 3.2.3 T_q shall be ≤ 0.05.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_q > 0.05.	A.1 Verify F_R^T is within the limits of LCO 3.2.2, "TOTAL RADIAL PEAKING FACTOR ".	2 hours <u>AND</u> Once per 8 hours thereafter
B. T_q > 0.10.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
C. Required Action and associated Completion Time not met. <u>OR</u> T_q > 0.15.	C.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify T_q is $\leq$ 0.05.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.4 AXIAL SHAPE INDEX (ASI)

LCO 3.2.4 The ASI shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ASI not within limits specified in COLR.	A.1 Restore ASI to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify ASI is within limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation

LCO 3.3.1 Four RPS trip units, associated instrument channels, and associated Zero Power Mode (ZPM) Bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Not applicable to High Startup Rate, Loss of Load, or ZPM Bypass Removal Functions. ---------------------------------- One or more Functions with one RPS trip unit or associated instrument channel inoperable.	A.1 Place affected trip unit in trip.	7 days
B. One High Startup Rate trip unit or associated instrument channel inoperable.	B.1 Restore trip unit and associated instrument channel to OPERABLE status.	Prior to entering MODE 2 from MODE 3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One Loss of Load trip unit or associated instrument channel inoperable.	C.1 Restore trip unit and associated instrument channel to OPERABLE status.	Prior to increasing THERMAL POWER to ≥ 17% RTP following entry into MODE 3
D. One or more ZPM Bypass Removal channels inoperable.	D.1 Remove the affected ZPM Bypasses.	Immediately
	OR	
	D.2 Declare affected trip units inoperable.	Immediately
E. -------------NOTE-------------- Not applicable to ZPM Bypass Removal Function. ----------------------------------- One or more Functions with two RPS trip units or associated instrument channels inoperable.	E.1 Place one trip unit in trip.	1 hour
	AND	
	-----------------NOTE------------------- Not applicable to High Startup Rate or Loss of Load Functions. --	
	E.2 Restore one trip unit and associated instrument channel to OPERABLE status.	7 days
F. Two power range channels inoperable.	F.1 Restrict THERMAL POWER to ≤ 70% RTP.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time not met. OR Control room ambient air temperature > 90ºF.	G.1 Be in MODE 3. AND G.2.1 Verify no more than one full-length control rod is capable of being withdrawn. OR G.2.2 Verify PCS boron concentration is at REFUELING BORON CONCENTRATION.	6 hours 6 hours 6 hours

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.1-1 to determine which SR shall be performed for each Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2	Verify control room temperature is ≤ 90°F.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	-----------------------------NOTE---------------------------- Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. --- Perform calibration (heat balance only) and adjust the power range excore and ΔT power channels to agree with calorimetric calculation if the absolute difference is ≥ 1.5%.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.4	-----------------------------NOTE---------------------------- Not required to be performed until 12 hours after THERMAL POWER is ≥ 25% RTP. --- Calibrate the power range excore channels using the incore detectors.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.5	Perform a CHANNEL FUNCTIONAL TEST and verify the Thermal Margin Monitor Constants.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.6	Perform a calibration check of the power range excore channels with a test signal.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.7	Perform a CHANNEL FUNCTIONAL TEST of High Startup Rate and Loss of Load Functions.	Once within 7 days prior to each reactor startup

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	-----------------------------NOTE----------------------------- Neutron detectors are excluded from the CHANNEL CALIBRATION. -- Perform a CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

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

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable High Power Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.8	≤ 15% RTP above current THERMAL POWER with a minimum of ≤ 30% RTP and a maximum of ≤ 109.4% RTP
2. High Startup Rate Trip[(b)]	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.8	NA
3. Low Primary Coolant System Flow Trip[(c)]	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 95%
4. Low Steam Generator A Level Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 25.9% narrow range
5. Low Steam Generator B Level Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 25.9% narrow range
6. Low Steam Generator A Pressure Trip[(c)]	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 500 psia
7. Low Steam Generator B Pressure Trip[(c)]	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≥ 500 psia
8. High Pressurizer Pressure Trip	1,2,3[(a)],4[(a)],5[(a)]	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.8	≤ 2255 psia

(a) With more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

(b) Trip may be bypassed when Wide Range Power is < 1E-4% RTP or when THERMAL POWER is > 13% RTP.

(c) Trips may be bypassed when Wide Range Power is < 1E-4% RTP. Bypass shall be automatically removed when Wide Range Power is ≥ 1E-4% RTP.

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

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Thermal Margin/ Low Pressure Trip[c]	1,2,3[a],4[a],5[a]	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.8	Table 3.3.1-2
10. Loss of Load Trip	1[d]	SR 3.3.1.7 SR 3.3.1.8	NA
11. Containment High Pressure Trip	1,2,3[a],4[a],5[a]	SR 3.3.1.5 SR 3.3.1.8	≤ 3.70 psig
12. Zero Power Mode Bypass Automatic Removal	1,2,3[a],4[a],5[a]	SR 3.3.1.8	NA

(a) With more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

(c) Trips may be bypassed when Wide Range Power is < 1E-4% RTP. Bypass shall be automatically removed when Wide Range Power is ≥ 1E-4% RTP.

(d) When THERMAL POWER is ≥ 17% RTP.

Table 3.3.1-2 (page 1 of 1)
Thermal Margin/Low Pressure Trip Function Allowable Value

The Allowable Value for the Thermal Margin/Low Pressure Trip, P_{trip}, is the higher of two values, P_{min} and P_{var}, both in psia:

$P_{min} = 1750$
$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9559$

Where:

QA = - 0.720(ASI) + 1.028;	when $- 0.628 \leq ASI < - 0.100$
QA = - 0.333(ASI) + 1.067;	when $- 0.100 \leq ASI < + 0.200$
QA = + 0.375(ASI) + 0.925;	when $+ 0.200 \leq ASI \leq + 0.565$
ASI = Measured ASI	when $Q \geq 0.0625$
ASI = 0.0	when $Q < 0.0625$
$QR_1 = 0.412(Q) + 0.588$;	when $Q \leq 1.0$
$QR_1 = Q$;	when $Q > 1.0$

Q = THERMAL POWER/RATED THERMAL POWER

T_{in} = Maximum primary coolant inlet temperature, in °F

ASI, T_{in}, and Q are the existing values as measured by the associated instrument channel.

3.3 INSTRUMENTATION

3.3.2 Reactor Protective System (RPS) Logic and Trip Initiation

LCO 3.3.2 Six channels of RPS Matrix Logic, four channels of RPS Trip Initiation Logic, and two channels of RPS Manual Trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, with more than one full-length control rod capable of being withdrawn and Primary Coolant System (PCS) boron concentration less than REFUELING BORON CONCENTRATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Matrix Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. One channel of Trip Initiation Logic inoperable.	B.1 De-energize the affected clutch power supplies.	1 hour
C. One channel of Manual Trip inoperable.	C.1 Restore channel to OPERABLE status.	Prior to entering MODE 2 from MODE 3
D. Two channels of Trip Initiation Logic affecting the same trip leg inoperable.	D.1 De-energize the affected clutch power supplies.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required Action and associated Completion Time not met. OR One or more Functions with two or more Manual Trip, Matrix Logic or Trip Initiation Logic channels inoperable for reasons other than Condition D.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2.1	Verify no more than one full-length control rod is capable of being withdrawn.	6 hours
	OR		
	E.2.2	Verify PCS boron concentration is at REFUELING BORON CONCENTRATION.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform a CHANNEL FUNCTIONAL TEST on each RPS Matrix Logic channel and each RPS Trip Initiation Logic channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2	Perform a CHANNEL FUNCTIONAL TEST on each RPS Manual Trip channel.	Once within 7 days prior to each reactor startup

3.3 INSTRUMENTATION

3.3.3 Engineered Safety Features (ESF) Instrumentation

LCO 3.3.3 Four ESF bistables and associated instrument channels for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: As specified in Table 3.3.3-1.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------------NOTE--------------- Not applicable to RAS. ------------------------------------- One or more Functions with one ESF bistable or associated instrument channel inoperable.	A.1 Place affected bistable in trip.	7 days
B. ---------------NOTE--------------- Not applicable to RAS. ------------------------------------- One or more Functions with two ESF bistables or associated instrument channels inoperable.	B.1 Place one bistable in trip. <u>AND</u> B.2 Restore one bistable and associated instrument channel to OPERABLE status.	8 hours 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One RAS bistable or associated instrument channel inoperable.	C.1 Bypass affected bistable. <u>AND</u> C.2 Restore bistable and associated instrument channel to OPERABLE status.	8 hours 7 days
D. Required Action and associated Completion Time not met for Functions 1, 2, 3, 4, or 7.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 30 hours
E. Required Action and associated Completion Time not met for Functions 5 or 6.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

---NOTE---

Refer to Table 3.3.3-1 to determine which SR shall be performed for each Function.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform a CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.3.2	Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.3.3	Perform a CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

Table 3.3.3-1 (page 1 of 2)
Engineered Safety Features Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection Signal (SIS)			
a. Pressurizer Low Pressure	1,2,3	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 1593 psia
2. Steam Generator Low Pressure Signal (SGLP)			
a. Steam Generator A Low Pressure	1,2[(a)],3[(a)]	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 500 psia
b. Steam Generator B Low Pressure	1,2[(a)],3[(a)]	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 500 psia
3. Recirculation Actuation Signal (RAS)			
a. SIRWT Low Level	1,2,3	SR 3.3.3.3	≥ 21 inches and ≤ 27 inches above tank bottom
4. Auxiliary Feedwater Actuation Signal (AFAS)			
a. Steam Generator A Low Level	1,2,3	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 25.9% narrow range
b. Steam Generator B Low Level	1,2,3	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 25.9% narrow range

(a) Not required to be OPERABLE when all Main Steam Isolation Valves (MSIVs) are closed and deactivated, and all Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves are either closed and deactivated, or isolated by closed manual valves.

Table 3.3.3-1 (page 2 of 2)
Engineered Safety Features Instrumentation

FUNCTION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Containment High Pressure (CHP)			
a. Containment High Pressure — Left Train	1,2,3,4	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 psig and ≤ 4.3 psig
b. Containment High Pressure — Right Train	1,2,3,4	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 psig and ≤ 4.3 psig
6. Containment High Radiation Signal (CHR)			
a. Containment High Radiation	1,2,3,4	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≤ 20 R/hour
7. Automatic Bypass Removals			
a. Pressurizer Low Pressure Bypass	1,2,3	SR 3.3.3.3	≤ 1700 psia
b. Steam Generator A Low Pressure Bypass	1,2[(a)],3[(a)]	SR 3.3.3.3	≤ 565 psia
c. Steam Generator B Low Pressure Bypass	1,2[(a)],3[(a)]	SR 3.3.3.3	≤ 565 psia

(a) Not required to be OPERABLE when all Main Steam Isolation Valves (MSIVs) are closed and deactivated, and all Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves are either closed and deactivated, or isolated by closed manual valves.

3.3 INSTRUMENTATION

3.3.4 Engineered Safety Features (ESF) Logic and Manual Initiation

LCO 3.3.4 Two ESF Manual Initiation and two ESF Actuation Logic channels and associated bypass removal channels shall be OPERABLE for each ESF Function specified in Table 3.3.4-1.

APPLICABILITY: According to Table 3.3.4-1.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one Manual Initiation, Bypass Removal, or Actuation Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. One or more Functions with two Manual Initiation, Bypass Removal, or Actuation Logic channels inoperable for Functions 1, 2, 3, or 4. OR Required Action and associated Completion Time of Condition A not met for Functions 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with two Manual Initiation, or Actuation Logic channels inoperable for Functions 5 or 6. OR Required Action and associated Completion Time of Condition A not met for Functions 5 or 6.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform functional test of each SIS actuation channel normal and standby power functions.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.2	Perform a CHANNEL FUNCTIONAL TEST of each AFAS actuation logic channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.3	Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.4-1 (page 1 of 1)
Engineered Safety Features Actuation Logic and Manual Initiation

	FUNCTION	APPLICABLE MODES
1.	Safety Injection Signal (SIS)[a]	1,2,3
2.	Steam Generator Low Pressure Signal (SGLP)[b][c]	1,2[d],3[d]
3.	Recirculation Actuation Signal (RAS)	1,2,3
4.	Auxiliary Feedwater Actuation Signal (AFAS)	1,2,3
5.	Containment High Pressure Signal (CHP)[c]	1,2,3,4
6.	Containment High Radiation Signal (CHR)	1,2,3,4

(a) SIS actuation by Pressurizer Low Pressure may be manually bypassed when pressurizer pressure is ≤ 1700 psia. The bypass shall be automatically removed whenever pressurizer pressure is > 1700 psia.

(b) SGLP actuation may be manually bypassed when SG pressure is ≤ 565 psia. The bypass shall be automatically removed whenever steam generator pressure is > 565 psia.

(c) Manual Initiation may be achieved by individual component controls.

(d) Not required to be OPERABLE when all Main Steam Isolation Valves (MSIVs) are closed and deactivated, and all Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves are either closed and deactivated, or isolated by closed manual valves.

3.3 INSTRUMENTATION

3.3.5 Diesel Generator (DG) - Undervoltage Start (UV Start)

LCO 3.3.5 Three channels of Loss of Voltage Function and three channels of Degraded Voltage Function auto-initiation instrumentation and associated logic channels for each DG shall be OPERABLE.

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel per DG inoperable.	A.1 Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG - UV Start instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform a CHANNEL FUNCTIONAL TEST on each DG-UV start logic channel.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.5.2	Perform CHANNEL CALIBRATION on each Loss of Voltage and Degraded Voltage channel with setpoints as follows: a. Degraded Voltage Function ≥ 2187 V and ≤ 2264 V 1. Time delay (degraded voltage sensing relay): ≥ 0.5 seconds and ≤ 0.8 seconds; and 2. Time delay (degraded voltage sensing relay plus time delay relay): ≥ 6.2 seconds and ≤ 7.1 seconds. b. Loss of Voltage Function ≥ 1780 V and ≤ 1940 V Time delay: ≥ 5.45 seconds and ≤ 8.15 seconds at 1400 V.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.6 Refueling Containment High Radiation (CHR) Instrumentation

LCO 3.3.6 Two Refueling CHR Automatic Actuation Function channels and two CHR Manual Actuation Function channels shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 Place the affected channel in trip.	4 hours
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	4 hours
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies within containment.	4 hours
B. One or more Functions with two channels inoperable.	B.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	B.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform a CHANNEL CHECK of each refueling CHR monitor channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.2	Perform a CHANNEL FUNCTIONAL TEST of each refueling CHR monitor channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.3	Perform a CHANNEL FUNCTIONAL TEST of each CHR Manual Initiation channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.4	Perform a CHANNEL CALIBRATION of each refueling CHR monitor channel.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.7 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.7 The PAM instrumentation for each Function in Table 3.3.7-1 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 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (Not Used)		
E. Required Action and associated Completion Time of Condition C not met.	E.1 Enter the Condition referenced in Table 3.3.7-1 for the channel.	Immediately
F. As required by Required Action E.1 and referenced in Table 3.3.7-1.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	6 hours 30 hours
G. As required by Required Action E.1 and referenced in Table 3.3.7-1.	G.1 Initiate action in accordance with Specification 5.6.6.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---
These SRs apply to each PAM instrumentation Function in Table 3.3.7-1.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2	-----------------------------NOTE----------------------------- Neutron detectors are excluded from the CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

Table 3.3.7-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION E.1
1.	Primary Coolant System Hot Leg Temperature (wide range)	2	F
2.	Primary Coolant System Cold Leg Temperature (wide range)	2	F
3.	Wide Range Neutron Flux	2	F
4.	Containment Floor Water Level (wide range)	2	F
5.	Subcooled Margin Monitor	2	F
6.	Pressurizer Level (wide range)	2	F
7.	(Deleted)		
8.	Condensate Storage Tank Level	2	F
9.	Primary Coolant System Pressure (wide range)	2	F
10.	Containment Pressure (wide range)	2	F
11.	Steam Generator A Water Level (wide range)	2	F
12.	Steam Generator B Water Level (wide range)	2	F
13.	Steam Generator A Pressure	2	F
14.	Steam Generator B Pressure	2	F
15.	Containment Isolation Valve Position	1 per valve[(a)]	F
16.	Core Exit Temperature - Quadrant 1	4	F
17.	Core Exit Temperature - Quadrant 2	4	F
18.	Core Exit Temperature - Quadrant 3	4	F
19.	Core Exit Temperature - Quadrant 4	4	F
20.	Reactor Vessel Water Level	2	G
21.	Containment Area Radiation (high range)	2	G

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

3.3 INSTRUMENTATION

3.3.8 Alternate Shutdown System

LCO 3.3.8 The Alternate Shutdown System Functions in Table 3.3.8-1 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 required Functions inoperable.	A.1 Restore required Functions 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 4.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL FUNCTIONAL TEST of the Source Range Neutron Flux Function.	Once within 7 days prior to each reactor startup
SR 3.3.8.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	In accordance with the Surveillance Frequency Control Program
SR 3.3.8.3	-----------------------------NOTES--------------------------- 1. Not required for Functions 16, 17, and 18. 2. Neutron detectors are excluded from the CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION for each required instrumentation channel.	In accordance with the Surveillance Frequency Control Program

Table 3.3.8-1 (page 1 of 1)
Alternate Shutdown System Instrumentation and Controls

	FUNCTION, INSTRUMENT OR CONTROL PARAMETER	REQUIRED CHANNELS
1.	Source Range Neutron Flux	1
2.	Pressurizer Pressure	1
3.	Pressurizer Level	1
4.	Primary Coolant System (PCS) #1 Hot Leg Temperature	1
5.	PCS #2 Hot Leg Temperature	1
6.	PCS #1 Cold Leg Temperature	1
7.	PCS #2 Cold Leg Temperature	1
8.	Steam Generator (SG) A Pressure	1
9.	SG B Pressure	1
10.	SG A Wide Range Level	1
11.	SG B Wide Range Level	1
12.	Safety Injection Refueling Water (SIRW) Tank Level	1
13.	Auxiliary Feedwater (AFW) Flow Indication to SG A	1
14.	AFW Flow Indication to SG B	1
15.	AFW Low Suction Pressure Alarm (P-8B)	1
16.	AFW Pump P-8B Steam Supply Valve Control	1
17.	AFW Flow Control to SG A	1
18.	AFW Flow Control to SG B	1

3.3 INSTRUMENTATION

3.3.9 Neutron Flux Monitoring Channels

LCO 3.3.9 Two channels of neutron flux monitoring instrumentation shall be OPERABLE.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channel(s) inoperable.	A.1	Suspend all operations involving positive reactivity additions.	Immediately
	<u>AND</u>		
	A.2	Perform SDM verification in accordance with SR 3.1.1.1.	4 hours <u>AND</u> Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.9.2	-----------------------------NOTE------------------------------ Neutron detectors are excluded from the CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.10 Engineered Safeguards Room Ventilation (ESRV) Instrumentation

LCO 3.3.10 Two channels of ESRV Instrumentation shall be OPERABLE.

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

ACTIONS

---NOTE---

Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Initiate action to isolate the associated ESRV System.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform a CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.10.2	Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.10.3	Perform a CHANNEL CALIBRATION. Verify high radiation setpoint on each ESRV Instrumentation radiation monitoring channel is ≤ 2.2E+5 cpm.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

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

LCO 3.4.1 PCS DNB parameters for pressurizer pressure, cold leg temperature, and PCS total flow rate shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer pressure, PCS cold leg temperature, or PCS total flow rate not within limits.	A.1 Restore 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	Verify pressurizer pressure within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2	Verify PCS cold leg temperature within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	-----------------------------NOTE--------------------------- Not required to be performed until 31 EFPD after THERMAL POWER is ≥ 90% RTP. -- Verify PCS total flow rate within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program <u>AND</u> After each plugging of 10 or more steam generator tubes

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.2 PCS Minimum Temperature for Criticality

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{ave} in one or more PCS 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 PCS T_{ave} in each loop ≥ 525ºF.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.3 PCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 PCS pressure, PCS temperature, and PCS heatup and cooldown rates shall be maintained within the limits of Figure 3.4.3-1 and Figure 3.4.3-2.

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. <u>AND</u> A.2 Determine PCS 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. <u>AND</u> B.2 Be in MODE 5 with PCS pressure < 270 psia.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. --------------NOTE------------- Required Action C.2 shall be completed whenever this Condition is entered. ---------------------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 AND C.2	Initiate action to restore parameter(s) to within limits. Determine PCS 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 PCS heatup and cooldown operations. -- Verify PCS pressure, PCS temperature, and PCS heatup and cooldown rates are within the limits of Figure 3.4.3-1 and Figure 3.4.3-2.	 In accordance with the Surveillance Frequency Control Program

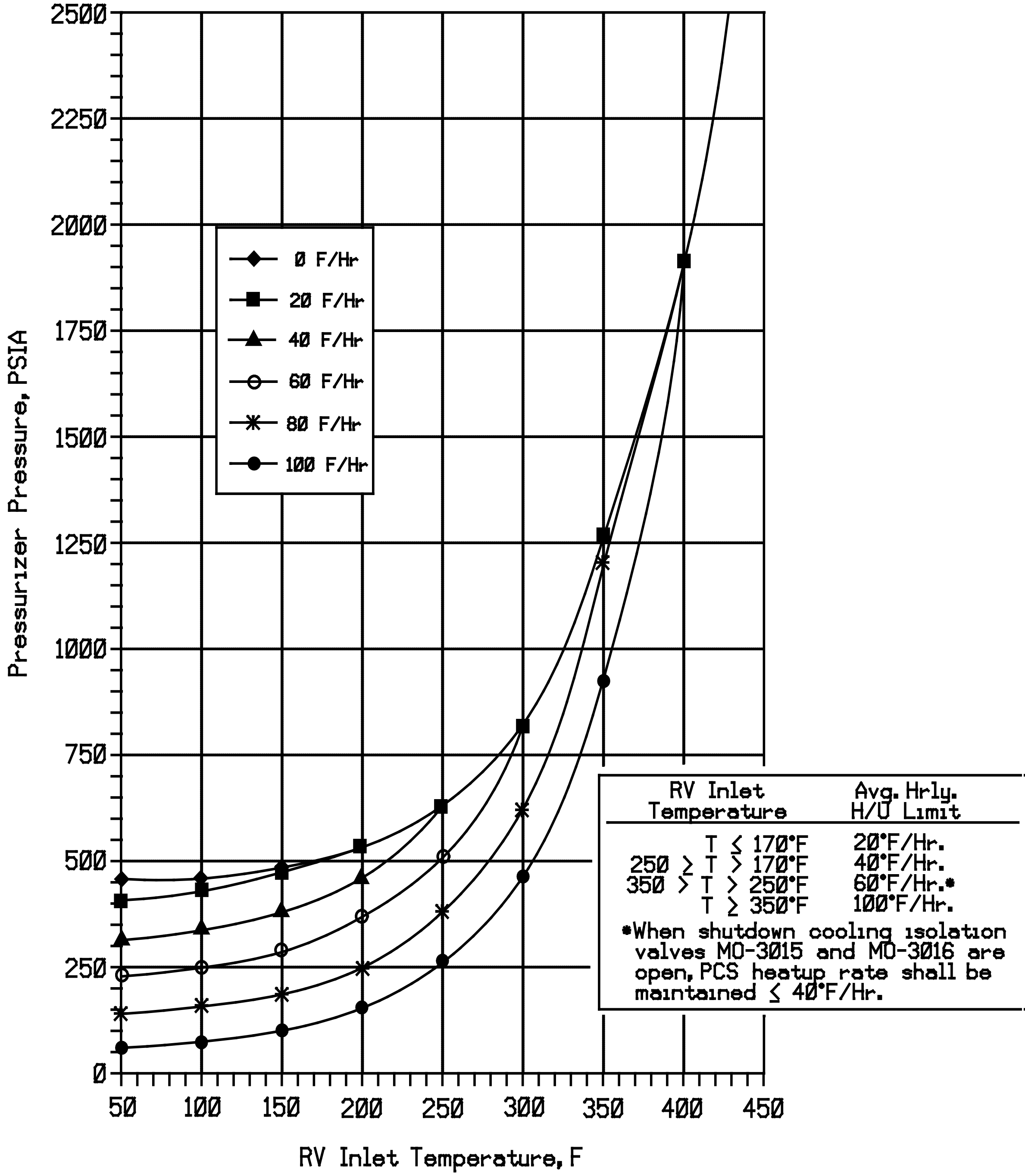
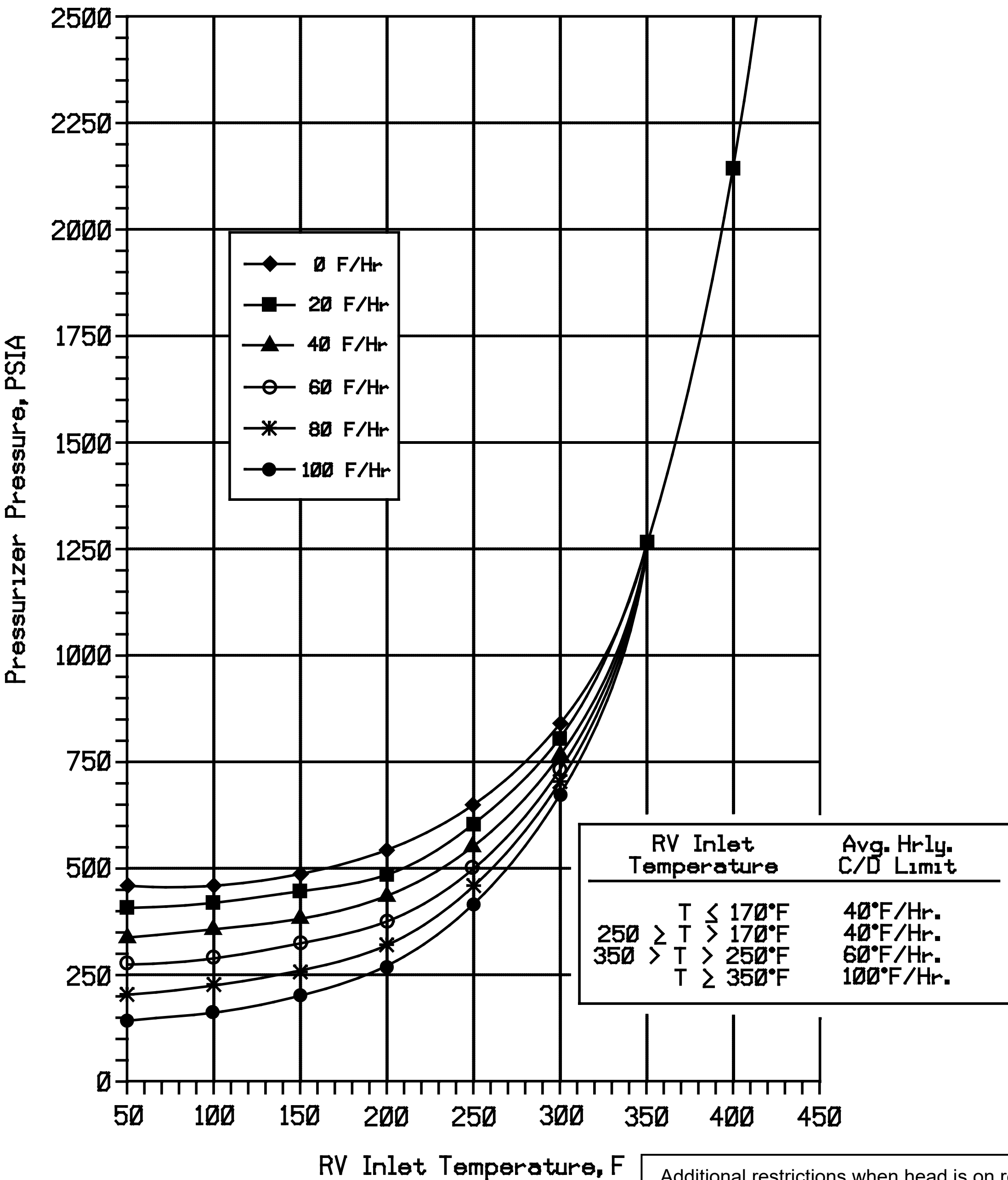


Figure 3.4.3-1 (Page 1 of 1)
Pressure – Temperature Limits for Heatups
Applicable up to 42.1 EFPY



Additional restrictions when head is on reactor vessel:

1. Maintain average core exit temperature:
   135°F ≥ T ≥ 110°F for ≥ 3 hours.

2. Following completion of item 1, maintain average hourly cooldown (C/D) limit of 20°F/hour based on core exit temperature indication.

Figure 3.4.3-2 (Page 1 of 1)
Pressure – Temperature Limits for Cooldown
Applicable up to 42.1 EFPY

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.4 PCS Loops - MODES 1 and 2

LCO 3.4.4 Two PCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each PCS loop is in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.5 PCS Loops - MODE 3

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

--NOTES--

1. All primary coolant pumps may not be in operation for ≤ 1 hour per 8 hour period, provided:

   a. No operations are permitted that would cause reduction of the PCS boron concentration; and

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

2. Forced circulation (starting the first primary coolant pump) shall not be initiated unless one of the following conditions is met:

   a. PCS cold leg temperature (T_c) is > 430°F;

   b. Steam Generator (SG) secondary temperature is equal to or less than the reactor inlet temperature (T_c);

   c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour; or

   d. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required PCS loop inoperable.	A.1 Restore required PCS loop to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	24 hours
C. No PCS loop OPERABLE. OR No PCS loop in operation.	C.1 Suspend all operations involving a reduction of PCS boron concentration. AND C.2 Initiate action to restore one PCS loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required PCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.2	Verify secondary side water level in each steam generator ≥ -84%.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.3	Verify correct breaker alignment and indicated power available to the required primary coolant pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.6 PCS Loops - MODE 4

LCO 3.4.6 Two loops or trains consisting of any combination of PCS loops and Shutdown Cooling (SDC) trains shall be OPERABLE, and either:

a. One PCS loop shall be in operation; or

b. One SDC train shall be in operation with ≥ 2810 gpm flow through the reactor core.

--NOTES--

1. All Primary Coolant Pumps (PCPs) and SDC pumps may not be in operation for ≤ 1 hour per 8 hour period, provided:

   a. No operations are permitted that would cause reduction of the PCS boron concentration; and

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

2. Forced circulation (starting the first PCP) shall not be initiated unless one of the following conditions is met:

   a. Steam Generator (SG) secondary temperature is equal to or less than the reactor inlet temperature (T_c),

   b. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour,

   c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

3. Primary coolant pumps P-50A and P-50B shall not be operated simultaneously.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PCS loop inoperable. <u>AND</u> Two SDC trains inoperable.	A.1 Initiate action to restore a second PCS loop or one SDC train to OPERABLE status.	Immediately
B. One SDC train inoperable. <u>AND</u> Two PCS loops inoperable.	B.1 Be in MODE 5.	24 hours
C. No PCS loops or SDC trains OPERABLE. <u>OR</u> No PCS loop in operation with SDC flow through the reactor core not within limits.	C.1 Suspend all operations involving reduction of PCS boron concentration.	Immediately
	<u>AND</u>	
	C.2.1 Initiate action to restore one PCS loop to OPERABLE status and operation.	Immediately
	<u>OR</u>	
	C.2.2 Initiate action to restore one SDC train to OPERABLE status and operation with $\geq$ 2810 gpm flow through the reactor core.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one SDC train is in operation with ≥ 2810 gpm flow through the reactor core, or one PCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	Verify secondary side water level in required SG(s) is ≥ -84%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.3	Verify correct breaker alignment and indicated power available to the required pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.7 PCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One Shutdown Cooling (SDC) train shall be OPERABLE and in operation with ≥ 2810 gpm flow through the reactor core, and either:

a. One additional SDC train shall be OPERABLE; or

b. The secondary side water level of each Steam Generator (SG) shall be ≥ -84%.

--NOTES--

1. The SDC pump of the train in operation may not be in operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the PCS boron concentration; and

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

2. Both SDC trains may be inoperable for up to 2 hours for surveillance testing or maintenance provided:

   a. One SDC train is providing the required flow through the reactor core;

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

   c. Each SG secondary side water level is ≥ -84%.

3. Forced circulation (starting the first primary coolant pump) shall not be initiated unless one of the following conditions is met:

   a. SG secondary temperature is equal to or less than the reactor inlet temperature (T_c);

   b. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour; or

   c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

4. Primary coolant pumps P-50A and P-50B shall not be operated simultaneously.

5. All SDC trains may not be in operation during planned heatup to MODE 4 when at least one PCS loop is in operation.

APPLICABILITY: MODE 5 with PCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SDC train inoperable. AND Any SG with secondary side water level not within limit.	A.1 Initiate action to restore a second SDC train to OPERABLE status. OR A.2 Initiate action to restore SG secondary side water levels to within limits.	Immediately Immediately
B. Two SDC trains inoperable. OR SDC flow through the reactor core not within limits.	B.1 Suspend all operations involving reduction in PCS boron concentration. AND B.2 Initiate action to restore one SDC train to OPERABLE status and operation with ≥ 2810 gpm flow through the reactor core.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one SDC train is in operation with ≥ 2810 gpm flow through the reactor core.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.2	Verify required SG secondary side water level is ≥ - 84%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.3	Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.8 PCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two Shutdown Cooling (SDC) trains shall be OPERABLE, and either:

a. One SDC train in operation with ≥ 2810 gpm flow through the reactor core; or

b. One SDC train in operation with ≥ 650 gpm flow through the reactor core with two of the three charging pumps incapable of reducing the boron concentration in the PCS below the minimum value necessary to maintain the required SHUTDOWN MARGIN.

---NOTES---

1. All SDC pumps may not be in operation for ≤ 1 hour provided:

   a. No operations are permitted that would cause a reduction of the PCS boron concentration;

   b. Core outlet temperature is maintained > 10°F below saturation temperature; and

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

2. One SDC train may be inoperable for ≤ 2 hours for surveillance testing provided the other SDC train is OPERABLE and in operation.

APPLICABILITY: MODE 5 with PCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SDC train inoperable.	A.1 Initiate action to restore SDC train to OPERABLE status.	Immediately
B. Two SDC trains inoperable. <u>OR</u> SDC flow through the reactor core not within limits.	B.1 Suspend all operations involving reduction of PCS boron concentration. <u>AND</u> B.2 Initiate action to restore one SDC train to OPERABLE status and operation with SDC flow through the reactor core within limit.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	-------------------------------NOTE-------------------------- Only required to be met when complying with LCO 3.4.8.a. --- Verify one SDC train is in operation with ≥ 2810 gpm flow through the reactor core.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.8.2	------------------------------NOTE------------------------- Only required to be met when complying with LCO 3.4.8.b. --- Verify one SDC train is in operation with ≥ 650 gpm flow through the reactor core.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.3	------------------------------NOTE------------------------- Only required to be met when complying with LCO 3.4.8.b. --- Verify two of three charging pumps are incapable of reducing the boron concentration in the PCS below the minimum value necessary to maintain the required SHUTDOWN MARGIN.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.4	Verify correct breaker alignment and indicated power available to the SDC pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level < 62.8%;

--NOTE--
The pressurizer water level limit does not apply in MODE 3 until after a bubble has been established in the pressurizer and the pressurizer water level has been lowered to within its normal operating band.
----

b. ≥ 375 kW of pressurizer heater capacity available from electrical bus 1D, and

c. ≥ 375 kW of pressurizer heater capacity available from electrical bus 1E with the capability of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor tripped. AND A.2 Be in MODE 4.	6 hours 30 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. < 375 kW pressurizer heater capacity available from electrical bus 1D, or electrical bus 1E, <u>OR</u> Required pressurizer heater capacity from electrical bus 1E not capable of being powered from an emergency power supply.	B.1 Restore required pressurizer heaters to OPERABLE status.	72 hours
C. ------------NOTE--------------- Not applicable when the remaining electrical bus 1D or electrical bus 1E required pressurizer heaters intentionally made inoperable. ----------------------------------- < 375 kW pressurizer heater capacity available from electrical bus 1D, and electrical bus 1E, <u>OR</u> < 375 kW pressurizer heater capacity available from electrical bus 1D, and required pressurizer heater capacity from electrical bus 1E not capable of being powered from an emergency power supply.	C.1 Restore at least electrical bus 1D or electrical bus 1E required pressurizer heaters to OPERABLE status.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-------------------------------NOTE--------------------------- Not required to be met until 1 hour after establishing a bubble in the pressurizer and the pressurizer water level has been lowered to within its normal operating band. --- Verify pressurizer water level is < 62.8%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify the capacity of pressurizer heaters from electrical bus 1D, and electrical bus 1E is ≥ 375 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.3	Verify the required pressurizer heater capacity from electrical bus 1E is capable of being powered from an emergency power supply.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings as specified in Table 3.4.10-1.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all PCS cold leg temperatures ≥ 430°F.

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 not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Reduce any PCS cold leg temperature < 430°F.	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% of required setpoint.	In accordance with the INSERVICE TESTING PROGRAM

Table 3.4.10-1 (page 1 of 1)
Pressurizer Safety Valve Lift Settings

VALVE NUMBER	LIFT SETTING (psia ± 3%)
RV-1039	2580
RV-1040	2540
RV-1041	2500

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

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

APPLICABILITY: MODES 1 and 2,
MODE 3 with all PCS cold leg temperatures ≥ 430°F.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PORV inoperable.	A.1 Close associated block valve. <u>AND</u> A.2 Restore PORV to OPERABLE status.	1 hour 72 hours
B. One block valve inoperable.	B.1 Place associated PORV in manual control. <u>AND</u> B.2 Restore block valve to OPERABLE status.	1 hour 72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two PORVs inoperable.	C.1 Close associated block valves. AND C.2 Restore at least one PORV to OPERABLE status.	1 hour 2 hours
D. Two block valves inoperable.	D.1 Place associated PORVs in manual control. AND D.2 Restore at least one block valve to OPERABLE status.	1 hour 2 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Perform a complete cycle of each block valve.	Once prior to entering MODE 4 from MODE 5 if not performed within previous 92 days
SR 3.4.11.2	Perform a complete cycle of each PORV with PCS average temperature > 200°F.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with:

a. Both High Pressure Safety Injection (HPSI) pumps incapable of injecting into the PCS, and

---NOTES---
1. LCO 3.4.12.a is only required when any PCS cold leg temperature is < 300°F.

2. LCO 3.4.12.a does not prohibit the use of the HPSI pumps for emergency addition of makeup to the PCS.
----

b. One of the following pressure relief capabilities:

   1. Two Power Operated Relief Valves (PORVs) with lift settings as specified in Figure 3.4.12-1; or

   2. The PCS depressurized and a PCS vent capable of relieving ≥ 167 gpm at a pressure of 315 psia.

APPLICABILITY: MODE 3 when any PCS cold leg temperature is < 430°F,
MODES 4 and 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

--NOTE--
LCO 3.0.4.b is not applicable to PORVs when entering MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two HPSI pumps capable of injecting into the PCS.	A.1 Initiate action to verify no HPSI pump is capable of injecting into the PCS.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required PORV inoperable and pressurizer water level ≤ 57%.	B.1 Restore required PORV to OPERABLE status.	7 days
C. One required PORV inoperable and pressurizer water level > 57%.	C.1 Restore required PORV to OPERABLE status.	24 hours
D. Two required PORVs inoperable. <u>OR</u> Required Action and associated Completion Time not met. <u>OR</u> LTOP System inoperable for any reason other than Condition A, B, or C.	D.1 Depressurize PCS and establish PCS vent capable of relieving ≥ 167 gpm at a PCS pressure of 315 psia.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	------------------------------NOTE------------------------ Only required to be met when complying with LCO 3.4.12.a. --- Verify both HPSI pumps are incapable of injecting into the PCS.	 In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify required PCS vent, capable of relieving $\geq$ 167 gpm at a PCS pressure of 315 psia, is open.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	Verify PORV block valve is open for each required PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	------------------------------NOTE------------------------ Not required to be performed until 12 hours after decreasing any PCS cold leg temperature to < 430°F. --- Perform CHANNEL FUNCTIONAL TEST on each required PORV, excluding actuation.	 In accordance with the Surveillance Frequency Control Program
SR 3.4.12.5	Perform CHANNEL CALIBRATION on each required PORV actuation channel.	In accordance with the Surveillance Frequency Control Program

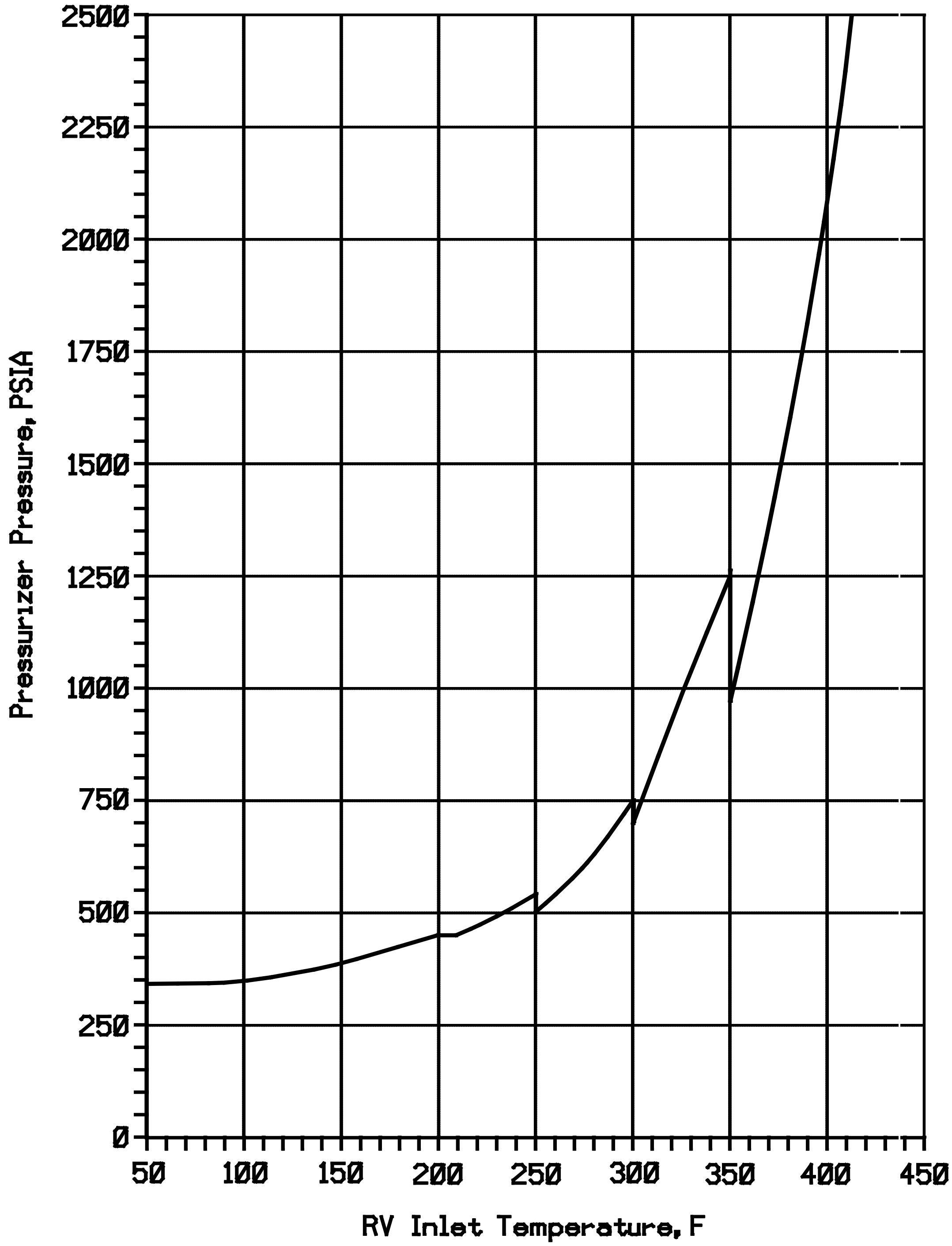


Figure 3.4.12-1 (Page 1 of 1)
LTOP Setpoint Limit
Applicable up to 42.1 EFPY

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.13 PCS Operational LEAKAGE

LCO 3.4.13 PCS 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. PCS 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 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 in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. -- Verify PCS operational LEAKAGE is within limits by performance of PCS water inventory balance.	----------NOTE-------- Only required to be performed during steady state operation -------------------------- In accordance with the Surveillance Frequency Control Program
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.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.14 PCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each PCS PIV shall be within limits and both Shutdown Cooling (SDC) suction valve interlocks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except during the SDC mode of operation, or transition to or from, the SDC mode of operation.

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 PCS PIVs not within limit.	------------------NOTE------------------ Each valve used to satisfy Required Action A.1 must have been verified to meet SR 3.4.14.1 and be on the PCS pressure boundary or the high pressure portion of the system. ---	(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	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
	AND		
	A.2	Restore PCS PIV to within limits.	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. One or both SDC suction valve interlocks inoperable.	C.1	Isolate the affected penetration by use of one closed deactivated valve.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-------------------------------NOTES-------------------------- 1. Only required to be performed in MODES 1 and 2. 2. Leakage rates ≤ 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible leakage rate of 5.0 gpm by 50% or greater. 3. Minimum test differential pressure shall not be less than 150 psid. --- Verify leakage from each PCS PIV is equivalent to ≤ 5 gpm at a PCS pressure of 2060 psia.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once prior to entering MODE 2 whenever the plant has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months
SR 3.4.14.2	Verify each SDC suction valve interlock prevents its associated valve from being opened with a simulated or actual PCS pressure signal ≥ 280 psia.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.3	------------------------------NOTE--------------------------- Only required to be performed in MODES 1 and 2. --- Verify each of the four Low Pressure Safety Injection (LPSI) check valves are closed.	Prior to entering MODE 2 after each use of the LPSI check valves for SDC

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.15 PCS Leakage Detection Instrumentation

LCO 3.4.15 Three of the following PCS leakage detection instrumentation channels shall be OPERABLE:

a. One containment sump level indicating channel;

b. One containment atmosphere gaseous activity monitoring channel;

c. One containment air cooler condensate level switch channel;

d. One containment atmosphere humidity monitoring channel.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required leak detection instrument channels inoperable.	A.1 Perform SR 3.4.13.1 (PCS water inventory balance).	Once per 24 hours
	AND	
	A.2 Restore inoperable channel(s) to OPERABLE status.	30 days
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. All required channels inoperable.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment sump level indicator.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform CHANNEL CHECK of the required containment atmosphere gaseous activity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CHECK of the required containment atmosphere humidity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL FUNCTIONAL TEST of the required containment air cooler condensate level switch.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.5	Perform CHANNEL CALIBRATION of the required containment sump level indicator.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.15.6	Perform CHANNEL CALIBRATION of the required containment atmosphere gaseous activity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.7	Perform CHANNEL CALIBRATION of the required containment atmosphere humidity monitor.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.16 PCS Specific Activity

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

APPLICABILITY: MODES 1 and 2,
MODE 3 with PCS average temperature (T_{ave}) ≥ 500°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 < 40 µCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 $\geq$ 40 μCi/gm. <u>OR</u> Gross specific activity of the primary coolant not within limit.	B.1 Be in MODE 3 with T_{ave} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify primary coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.2	-------------------------------NOTE--------------------------- Only required to be performed in MODE 1. -- Verify primary coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Once between 2 and 6 hours after THERMAL POWER change of ≥ 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 $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	In accordance with the Surveillance Frequency Control Program

3.4 PRIMARY COOLANT SYSTEM (PCS)

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 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 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. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days 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. AND B.2 Be in MODE 5.	6 hours 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 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 Safety Injection Tanks (SITs)

LCO 3.5.1 Four SITs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SIT inoperable due to boron concentration not within limits. <u>OR</u> One SIT inoperable due to the inability to verify level or pressure.	A.1 Restore SIT to OPERABLE status.	72 hours
B. One SIT inoperable for reasons other than Condition A.	B.1 Restore SIT to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
D. Two or more SITs inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each SIT isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each SIT is $\geq$ 1040 ft^3 and $\leq$ 1176 ft^3.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is $\geq$ 200 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each SIT is $\geq$ 1720 ppm and $\leq$ 2500 ppm.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.5	Verify power is removed from each SIT isolation valve operator.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Primary Coolant System (PCS) temperature ≥ 325°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LPSI subsystem inoperable.	A.1 Restore LPSI subsystem to OPERABLE status.	7 days
B. One or more ECCS 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 Reduce PCS temperature to < 325°F.	6 hours 24 hours
D. Less than 100% of the required ECCS flow available.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves and hand switches are in the open position. <u>Valve/Hand Switch Number</u> — <u>Function</u> CV-3027 — SIRWT Recirc Valve HS-3027A — Hand Switch For CV-3027 HS-3027B — Hand Switch For CV-3027 CV-3056 — SIRWT Recirc Valve HS-3056A — Hand Switch For CV-3056 HS-3056B — Hand Switch For CV-3056	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.2	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.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify CV-3006, "SDC Flow Control Valve," is open and its air supply is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	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 3.5.2.5	Verify each ECCS automatic valve that is not locked, sealed, or otherwise secured in position, in the flow path actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.7	Verify each LPSI pump stops on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.8	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. <u>Valve Number</u> — <u>Function</u> MO-3008 — LPSI to Cold leg 1A MO-3010 — LPSI to Cold leg 1B MO-3012 — LPSI to Cold leg 2A MO-3014 — LPSI to Cold leg 2B MO-3082 — HPSI to Hot leg 1 MO-3083 — HPSI to Hot leg 1	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.9	Verify, by visual inspection, the containment sump passive strainer assemblies are not restricted by debris, and the containment sump passive strainer assemblies and other containment sump entrance pathways show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 One Low Pressure Safety Injection (LPSI) train shall be OPERABLE.

---NOTE---
A LPSI train may be considered OPERABLE during alignment and operation for shutdown cooling if capable of being manually realigned to the ECCS mode of operation.

APPLICABILITY: MODE 3 with Primary Coolant System (PCS) temperature < 325°F,
MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required LPSI train inoperable.	A.1 Initiate action to restore one LPSI train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs of Specification 3.5.2, "ECCS - Operating," are applicable: SR 3.5.2.2 SR 3.5.2.9 SR 3.5.2.4	In accordance with applicable SRs

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Safety Injection Refueling Water Tank (SIRWT)

LCO 3.5.4 The SIRWT shall be OPERABLE.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify SIRWT borated water temperature is ≥ 40°F and ≤ 100°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	--------------------------------NOTE-------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify SIRWT borated water volume is ≥ 250,000 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	--------------------------------NOTE-------------------------- Only required to be met in MODE 4. --- Verify SIRWT borated water volume is ≥ 200,000 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.4	Verify SIRWT boron concentration is ≥ 1720 ppm and ≤ 2500 ppm.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Containment Sump Buffering Agent and Weight Requirements

LCO 3.5.5 Buffer baskets shall contain ≥ 8,186 lbs and ≤10,553 lbs of Sodium Tetraborate Decahydrate (STB) $Na_2B_4O_7 \cdot 10H_2O$.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. STB not within limits.	A.1 Restore STB to within limits.	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 4.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	Verify the STB baskets contain ≥ 8,186 lbs and ≤10,553 lbs of equivalent weight sodium tetraborate decahydrate.	In accordance with the Surveillance Frequency Control Program
SR 3.5.5.2	Verify that a sample from the STB baskets provides adequate pH adjustment of borated water.	In accordance with the Surveillance Frequency Control Program

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 Leak Rate Testing Program.	In accordance with the Containment Leak Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Structural Integrity Surveillance Program.	In accordance with the Containment Structural Integrity Surveillance 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 through a “locked” air lock door 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 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.	1 hour
	<u>AND</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed in the affected air lock. AND ------------------NOTE----------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --- A.3 Verify the OPERABLE door is locked closed in the affected air lock.	24 hours 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. AND	 1 hour (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Lock an OPERABLE door closed in the affected air lock. <u>AND</u> ------------------NOTE----------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --- B.3 Verify an OPERABLE door is locked closed in the affected air lock.	24 hours 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. <u>AND</u> C.2 Verify a door is closed in the affected air lock. <u>AND</u> C.3 Restore air lock to OPERABLE status.	Immediately 1 hour 24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 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 Leak Rate Testing Program.	In accordance with the Containment Leak Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	In accordance with the Surveillance Frequency Control Program

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 8 inch purge exhaust valves and 12 inch air room supply valves 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 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 containment isolation valves. ----------------------------------- One or more penetration flow paths with one containment isolation valve inoperable (except for purge exhaust valve or air room supply valve not locked closed).	A.1 AND	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.	4 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	--------------------NOTE---------------- Isolation devices in high radiation areas may be verified by use of administrative means. --- A.2 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 containment isolation valves. ----------------------------------- One or more penetration flow paths with two containment isolation valves inoperable (except for purge exhaust valve or air room supply valve not locked closed).	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 and a closed system. ---------------------------------- 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. <u>AND</u> -------------------NOTE----------------- Isolation devices in high radiation areas may be verified by use of administrative means. -- C.2 Verify the affected penetration flow path is isolated.	72 hours Once per 31 days
D. One or more purge exhaust or air room supply valves not locked closed.	D.1 Lock closed the affected valves.	1 hour
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 8 inch purge valve and 12 inch air room supply valve is locked closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	-----------------------------NOTE---------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --	
	Verify each manual containment isolation valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured in position, and is required to be closed during accident conditions, is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	----------------------------NOTE----------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --	
	Verify each manual containment isolation valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured in position, 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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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
SR 3.6.3.5	Verify each containment 8 inch purge exhaust and 12 inch air room supply valve is closed by performance of a leakage rate test.	In accordance with the Surveillance Frequency Control Program
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.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≤ 1.0 psig in MODES 1 and 2 and ≤ 1.5 psig in MODES 3 and 4.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limit.	A.1 Restore containment pressure to within limit.	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 3.6.4.1	Verify containment pressure is within limit.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 140°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.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Cooling Systems

LCO 3.6.6 Two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment cooling trains inoperable.	A.1 Restore train(s) to OPERABLE status.	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 4.	6 hours 30 hours
C. Less than 100% of the required post accident containment cooling capability available.	C.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.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2	Operate each Containment Air Cooler Fan Unit for $\geq$ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.3	Verify the containment spray piping is full of water to the 735 ft elevation in the containment spray header.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify total service water flow rate, when aligned for accident conditions, is $\geq$ 4800 gpm to Containment Air Coolers VHX-1, VHX-2, and VHX-3.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.5	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 3.6.6.6	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to its correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.7	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.8	Verify each containment cooling fan starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.9	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

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

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Restore required MSSVs to OPERABLE status.	4 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 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each required MSSV lift setting is within the limits of 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

VALVE NUMBER		LIFT SETTING (psig ± 3%)
Steam Generator A	Steam Generator B	
RV-0703 RV-0704 RV-0705 RV-0706	RV-0701 RV-0702 RV-0707 RV-0708	1025
RV-0713 RV-0714 RV-0715 RV-0716	RV-0709 RV-0710 RV-0711 RV-0712	1005
RV-0717 RV-0718 RV-0723 RV-0724	RV-0719 RV-0720 RV-0721 RV-0722	985

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 both MSIVs are closed and de-activated.

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. AND C.2 Verify MSIV is closed.	8 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify closure time of each MSIV is ≤ 5 seconds on an actual or simulated actuation signal from each train under no flow conditions.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

LCO 3.7.3 Two MFRVs and two MFRV bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when both MFRVs and both MFRV bypass valves are either closed and de-activated, or isolated by closed manually actuated valves.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFRVs or MFRV bypass valves inoperable.	A.1 Close or isolate inoperable MFRV(s) or MFRV bypass valve(s).	8 hours
	<u>AND</u>	
	A.2 Verify inoperable MFRV(s) or MFRV bypass valve(s) is closed or isolated.	Once per 7 days
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.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFRV and MFRV bypass valve is ≤ 22 seconds on a actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valves (ADVs)

LCO 3.7.4 One ADV per steam generator shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADV inoperable.	A.1 Restore ADV to OPERABLE status.	7 days
B. Two required ADVs inoperable.	B.1 Restore one ADV to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ADV.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Two AFW trains shall be OPERABLE.

---NOTES---

1. Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

2. The steam driven pump is only required to be OPERABLE prior to making the reactor critical.

3. Two AFW pumps may be placed in manual for testing, for a period of up to 4 hours.

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.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AFW trains inoperable in MODE 1, 2, or 3.	A.1 Restore train(s) to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Less than 100% of the required AFW flow available to either steam generator. OR Less than two AFW pumps OPERABLE in MODE 1, 2, OR 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours
C. Less than 100% of the required AFW flow available, to both steam generators.	------------------NOTE------------------ LCO 3.0.3 and all other LCO Required Actions requiring MODE changes or power reductions are suspended until at least 100% of the required AFW flow is available. --- C.1 Initiate action to restore one AFW train to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each required AFW manual, power operated, and automatic valve in each water flow path and in the steam supply flow path to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	------------------------NOTE------------------------------- Not required to be met for the turbine driven AFW pump in MODE 3 below 800 psig in the steam generators. --- Verify the developed head of each required AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	---------------------------NOTE--------------------------- Only required to be met in MODES 1, 2 or 3 when AFW is not in operation. --- 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.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	----------------------------NOTE--------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify each required AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage and Supply

LCO 3.7.6 The combined useable volume of the Condensate Storage Tank (CST) and Primary Makeup Storage Tank (T-81) shall be ≥ 100,000 gallons.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Condensate volume not within limit.	A.1 Verify OPERABILITY of backup water supplies. AND A.2 Restore condensate volume to within limit.	4 hours AND Once per 12 hours thereafter 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance on steam generator for heat removal.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify condensate useable volume is ≥ 100,000 gallons.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more CCW trains inoperable.	A.1 Restore train(s) to OPERABLE status.	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. Less than 100% of the required post accident CCW cooling capability available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----------------------------NOTE--------------------------- Isolation of CCW flow to individual components does not render the 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.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	-----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, and 3. --- 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.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	-----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify each CCW pump starts automatically on an actual or simulated actuation signal in the “with standby power available” mode.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SWS trains inoperable.	A.1 Restore train(s) to OPERABLE status.	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. Less than 100% of the required post accident SWS cooling capability available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	----------------------------NOTE---------------------------- Isolation of SWS flow to individual components does not render 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.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, and 3. --- 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.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify each SWS pump starts automatically on an actual or simulated actuation signal in the “with standby power available” mode.	In accordance with the Surveillance Frequency Control Program

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 ≥ 568.25 ft above mean sea level.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.2	Verify water temperature of UHS is ≤ 85°F.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.10 Control Room Ventilation (CRV) Filtration

LCO 3.7.10 Two CRV Filtration trains shall be OPERABLE.

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

APPLICABILITY: MODES 1, 2, 3, 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies,
During movement of a fuel cask in or over the Spent Fuel Pool (SFP).

ACTIONS

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE------------ Not applicable when second CRV Filtration train intentionally made inoperable. --------------------------------- Two CRV Filtration trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>	
	C.2 Verify LCO 3.4.16, “PCS Specific Activity,” is met.	1 hour
	<u>AND</u>	
	C.3 Restore at least one CRV Filtration train to OPERABLE status.	24 hours
D. Required Action and associated Completion Time of Condition A not met during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP.	D.1 Place OPERABLE CRV Filtration train in emergency mode.	Immediately
	<u>OR</u>	
	D.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	D.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	D.2.3 Suspend movement of a fuel cask in or over the SFP.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CRV Filtration trains inoperable during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP. <u>OR</u> One or more CRV Filtration trains inoperable due to an inoperable CRE boundary during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP.	E.1 Suspend CORE ALTERATIONS. <u>AND</u> E.2 Suspend movement of irradiated fuel assemblies. <u>AND</u> E.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately
F. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRV Filtration train for ≥ 10 continuous hours with associated heater (VHX-26A or VHX-26B) operating.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required CRV Filtration filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.10.3	-----------------------------NOTE----------------------------- Only required to be met in MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies in containment. -- Verify each CRV Filtration train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
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

3.7 PLANT SYSTEMS

3.7.11 Control Room Ventilation (CRV) Cooling

LCO 3.7.11 Two CRV Cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies,
During movement of a fuel cask in or over the Spent Fuel Pool (SFP).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRV Cooling train inoperable.	A.1 Restore CRV Cooling train to OPERABLE status.	30 days
B. -------------NOTE-------------- Not applicable when second CRV Cooling train intentionally made inoperable. ----------------------------------- Two CRV Cooling trains inoperable in MODE 1, 2, 3, or 4.	B.1 Restore at least one CRV Cooling train to OPERABLE status.	24 hours

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 not met during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or movement of a fuel cask in or over the SFP.	D.1 Place OPERABLE CRV Cooling train in operation. OR D.2.1 Suspend CORE ALTERATIONS. AND D.2.2 Suspend movement of irradiated fuel assemblies. AND D.2.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CRV Cooling trains inoperable during CORE ALTERATIONS, during movement of irradiated fuel assemblies, or movement of a fuel cask in or over the SFP.	E.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	E.3 Suspend movement of a fuel cask in or over the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CRV Cooling train has the capability to remove the assumed heat load.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.12 Fuel Handling Area Ventilation System

LCO 3.7.12 The Fuel Handling Area Ventilation System shall be OPERABLE with one fuel handling area exhaust fan aligned to the emergency filter bank and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building,

During movement of a fuel cask in or over the SFP when irradiated fuel assemblies with < 90 days decay time are in the fuel handling building,

During CORE ALTERATIONS when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open,

During movement of irradiated fuel assemblies in the containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel Handling Area Ventilation System not aligned or in operation. OR Fuel Handling Area Ventilation System inoperable.	A.1 Suspend movement of fuel assemblies. AND A.2 Suspend CORE ALTERATIONS. AND A.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Perform required Fuel Handling Area Ventilation System filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.12.2	Verify the flow rate of the Fuel Handling Area Ventilation System, when aligned to the emergency filter bank, is ≥ 5840 cfm and ≤ 8760 cfm.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

LCO 3.7.13 Two ESRV Damper trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESRV Damper trains inoperable.	A.1 Initiate action to isolate associated ESRV Damper train(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify each ESRV Damper train closes on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool (SFP) Water Level

LCO 3.7.14 The SFP water level shall be ≥ 647 ft elevation.

-----NOTE-----
SFP level may be below the 647 ft elevation to support fuel cask movement, if the displacement of water by the fuel cask when submerged in the SFP, would raise SFP level to ≥ 647 ft elevation.

APPLICABILITY: During movement of irradiated fuel assemblies in the SFP,
During movement of a fuel cask in or over the SFP.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in SFP.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of fuel cask in or over the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the SFP water level is ≥ 647 ft elevation.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.15 The SFP boron concentration shall be ≥ 1720 ppm.

APPLICABILITY: When fuel assemblies are stored in the Spent Fuel Pool.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the SFP.	Immediately
	AND	
	A.2 Initiate action to restore SFP boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the SFP boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Storage

LCO 3.7.16 Storage in the spent fuel pool shall be as follows:

a. Each fuel assembly and non-fissile bearing component stored in a Region I Carborundum equipped storage rack shall be within the limitations in Specification 4.3.1.1 and, as applicable, within the requirements of the maximum nominal planar average U-235 enrichment and burnup of Tables 3.7.16-2, 3.7.16-3, 3.7.16-4 or 3.7.16-5,

b. Fuel assemblies in a Region I Metamic equipped storage rack shall be within the limitations in Specification 4.3.1.2, and

c. The combination of maximum nominal planar average U-235 enrichment, burnup, and decay time of each fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly or non-fissile bearing component is stored in the spent fuel pool or the north tilt pit.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to restore the noncomplying fuel assembly or non-fissile bearing component within requirements.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means each fuel assembly or non-fissile bearing component meets fuel storage requirements.	Prior to storing the fuel assembly or non-fissile bearing component in the spent fuel pool

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. <u>AND</u> 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 within limit.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

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

b. Two Diesel Generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

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

ACTIONS

--NOTE--

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 (offsite source check) for OPERABLE offsite circuit. AND A.2 Restore offsite circuit to OPERABLE status.	1 hour AND Once per 8 hours thereafter 72 hours AND 10 days from discovery of failure to meet LCO

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 (offsite source check) for the OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG 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 DG is not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 (start test) for OPERABLE DG.	24 hours
	AND		
	B.4	Restore DG to OPERABLE status.	7 days AND 10 days from discovery of failure to meet LCO

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. AND C.2 Restore one offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours
D. One offsite circuit inoperable. AND One DG 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. OR D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG 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. Three or more AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and voltage for each offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	Verify each DG starts from standby conditions and achieves: a. In ≤ 10 seconds, ready-to-load status; and b. Steady state voltage ≥ 2280 V and ≤ 2520 V, and frequency ≥ 59.5 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-----------------------------NOTES--------------------------- 1. Momentary transients outside the load range do not invalidate this test. 2. This Surveillance shall be conducted on only one DG at a time. 3. This Surveillance shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2. --- Verify each DG is synchronized and loaded, and operates for ≥ 60 minutes: a. For ≥ 15 minutes loaded to greater than or equal to peak accident load; and b. For the remainder of the test at a load ≥ 2300 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify each day tank contains ≥ 2500 gallons of fuel oil.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 68 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 2280 V and ≤ 2640 V; and c. Within 3 seconds following load rejection, the frequency is ≥ 59.5 Hz and ≤ 61.5 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.6	Verify each DG, operating at a power factor ≤ 0.9, does not trip, and voltage is maintained ≤ 4000 V during and following a load rejection of ≥ 2300 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.7	----------------------------NOTE---------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage ≥ 2280 V and ≤ 2520 V, 4. maintains steady state frequency ≥ 59.5 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	----------------------------NOTE---------------------------- Momentary transients outside the load and power factor ranges do not invalidate this test. -- Verify each DG, operating at a power factor ≤ 0.9, operates for ≥ 24 hours: a. For ≥ 100 minutes loaded ≥ its peak accident loading; and b. For the remaining hours of the test loaded ≥ 2300 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.9	----------------------------NOTE---------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify each DG: a. Synchronizes with offsite power source while supplying its associated 2400 V bus upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	------------------------------NOTE------------------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify the time of each sequenced load is within ± 0.3 seconds of design timing for each automatic load sequencer.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.11	------------------------------NOTE------------------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated safety injection signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through its automatic load sequencer, 3. achieves steady state voltage ≥ 2280 V and ≤ 2520 V, 4. achieves steady state frequency ≥ 59.5 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

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 Diesel Generator (DG) 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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The 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.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies. AND A.2.3 Initiate action to suspend operations involving positive reactivity additions. AND A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately Immediately Immediately
B. The required DG inoperable.	B.1 Suspend CORE ALTERATIONS. AND B.2 Suspend movement of irradiated fuel assemblies. AND B.3 Initiate action to suspend operations involving positive reactivity additions. AND B.4 Initiate action to restore required DG to OPERABLE status.	Immediately Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	For AC sources required to be OPERABLE, the following SRs of Specification 3.8.1, "AC Sources - Operating" are applicable: SR 3.8.1.1 SR 3.8.1.2 SR 3.8.1.4.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel, Lube Oil, and Starting Air

LCO 3.8.3 For each Diesel Generator (DG):

a. The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits, and

b. Both diesel fuel oil transfer systems shall be OPERABLE.

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel oil inventory less than a 7 day supply and greater than a 6 day supply.	A.1 Restore fuel oil inventory to within limits.	48 hours
B. Stored lube oil inventory less than a 7 day supply and greater than a 6 day supply.	B.1 Restore stored lube oil inventory to within limits.	48 hours
C. Fuel transfer system (P-18A) inoperable.	C.1 Restore fuel transfer system to OPERABLE status.	12 hours
D. Fuel transfer system (P-18B) inoperable.	D.1 Restore fuel transfer system to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Both fuel transfer systems inoperable.	E.1 Restore one fuel transfer system to OPERABLE status.	8 hours
F. Fuel oil properties other than viscosity, and water and sediment, not within limits.	F.1 Restore stored fuel oil properties to within limits.	30 days
G. Required Action and associated Completion Time not met. <u>OR</u> Stored diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, or F.	G.1 Declare associated DG(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify the fuel oil storage subsystem contains ≥ a 7 day supply of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	Verify stored lube oil inventory is ≥ a 7 day supply.	In accordance with the Surveillance Frequency Control Program
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 Fuel Oil Testing Program.	In accordance with the Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ 200 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove excess accumulated water from the fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.6	Verify the fuel oil transfer system operates to transfer fuel oil from the fuel oil storage tank to each DG day tank and engine mounted tank.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Left Train and Right Train DC electrical power sources shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required DC electrical power source battery charger inoperable.	A.1	Verify functional cross-connected battery charger is connected supplying power to the affected DC train.	2 hours
	<u>AND</u>		
	A.2	Restore required DC electrical power source battery charger to OPERABLE status.	7 days
B. One required DC electrical power source battery inoperable.	B.1	Verify OPERABLE directly connected and functional cross-connected battery chargers are connected supplying power to the affected DC train.	2 hours
	<u>AND</u>		
	B.2	Restore required DC electrical power source battery to OPERABLE status.	24 hours

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 125 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. <u>OR</u> Verify battery connection resistance is ≤ 50 μohm for inter-cell connections, ≤ 360 μohm for inter-rack connections, and ≤ 360 μohm for inter-tier connections.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	Inspect battery cells, cell plates, and racks for visual indication of physical damage or abnormal deterioration that could degrade battery performance.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.4	Remove visible terminal corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Verify battery connection resistance is ≤ 50 μohm for inter-cell connections, ≤ 360 μohm for inter-rack connections, and ≤ 360 μohm for inter-tier connections.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.6	Verify each required battery charger supplies ≥ 180 amps at ≥ 125 V for ≥ 8 hours.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.7	-----------------------------NOTES-------------------------- 1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8	-------------------------------NOTE--------------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program 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

3.8.5 DC Sources - Shutdown

LCO 3.8.5 DC electrical power source(s) shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems — Shutdown."

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power sources inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Initiate action to restore required DC electrical power source(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.3 SR 3.8.4.5 SR 3.8.4.2 SR 3.8.4.4 SR 3.8.4.6.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Left Train and Right Train batteries shall be within limits.

APPLICABILITY: When associated DC electrical power source(s) 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 cell parameters not within Category A or B limits.	A.1 Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	AND	
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours AND Once per 7 days thereafter
	AND	
	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells < 70°F. <u>OR</u> One or more batteries with one or more battery cell parameters not within Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.2	Verify average electrolyte temperature of representative cells is ≥ 70°F.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.3	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	In accordance with the Surveillance Frequency Control Program

Table 3.8.6-1 (page 1 of 1)
Battery Surveillance Requirements

PARAMETER	CATEGORY A: NORMAL LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: NORMAL LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[(b)(c)]	≥ 1.205	≥ 1.200 <u>AND</u> Average of connected cells ≥ 1.205	Not more than 0.020 below average connected cells <u>AND</u> Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 Four inverters shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One inverter inoperable.	-----------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any Preferred AC bus de-energized. --- A.1 Restore inverter to OPERABLE status.	 24 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 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to Preferred AC buses.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 Inverter(s) shall be OPERABLE to support the onsite Class 1E Preferred AC bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems — Shutdown."

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage, frequency, and alignment to required Preferred AC buses.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Left Train and Right Train AC, DC, and Preferred AC 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 in one train inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
B. One Preferred AC bus inoperable.	B.1 Restore Preferred AC bus to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
C. One or more DC electrical power distribution subsystems in one train inoperable.	C.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours
E. Two or more inoperable distribution subsystems 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 Preferred AC bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and Preferred AC 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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or Preferred AC bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate actions to restore required AC, DC, and Preferred AC bus electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.5 Declare associated required shutdown cooling train 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 Preferred AC bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Primary Coolant System and the refueling cavity shall be maintained at the REFUELING BORON CONCENTRATION.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is at the REFUELING BORON CONCENTRATION.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range channels shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range channel inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. Two source range channels inoperable.	B.1 Initiate action to restore one source range channel to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1 (PCS boron concentration verification).	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.9.2.2	-----------------------------NOTE------------------------------ Neutron detectors are excluded from the CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

--NOTE---
The equipment hatch is only required to be closed when the Fuel Handling Area Ventilation System is not in compliance with LCO 3.7.12, "Fuel Handling Area Ventilation System."
----

b. One door in the personnel air lock closed;

---NOTE---
One door in the personnel air lock is only required to be closed when the equipment hatch is closed.
----

c. One door in the emergency air lock closed; and

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Refueling Containment High Radiation Initiation signal.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required to be met containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.2	-----------------------------NOTE----------------------------- Only required to be met for unisolated containment penetrations. --- Verify each required automatic isolation valve closes on an actual or simulated Refueling Containment High Radiation signal.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.4 Shutdown Cooling (SDC) and Coolant Circulation - High Water Level

LCO 3.9.4 One SDC train shall be OPERABLE and in operation.

---NOTES---

1. The required SDC train may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Primary Coolant System boron concentration.

2. The required SDC train may be made inoperable for ≤ 2 hours per 8 hour period for testing or maintenance, provided one SDC train is in operation providing flow through the reactor core, and core outlet temperature is ≤ 200°F.

APPLICABILITY: MODE 6 with the refueling cavity water level ≥ 647 ft elevation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required SDC train inoperable or not in operation.	A.1 Initiate action to restore SDC train to OPERABLE status and operation. <u>AND</u> A.2 Suspend operations involving a reduction in primary coolant boron concentration. <u>AND</u>	Immediately Immediately (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND		
	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one SDC train is in operation and circulating primary coolant at a flow rate of ≥ 1000 gpm.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.5 Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level

LCO 3.9.5 Two SDC trains shall be OPERABLE, and one SDC train shall be in operation.

APPLICABILITY: MODE 6 with the refueling cavity water level < 647 ft elevation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SDC train inoperable.	A.1 Initiate action to restore SDC train to OPERABLE status.	Immediately
	OR	
	A.2 Initiate action to establish the refueling cavity water level ≥ 647 ft elevation.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No SDC train OPERABLE or in operation.	B.1 Suspend operations involving a reduction in primary coolant boron concentration.	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore one SDC train to OPERABLE status and to operation.	Immediately
	<u>AND</u>	
	B.3 Initiate action to close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one SDC train is in operation and circulating primary coolant at a flow rate of ≥ 1000 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LCO 3.9.6 The refueling cavity water level shall be maintained ≥ 647 ft elevation.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling cavity water level is ≥ 647 ft elevation.	In accordance with the Surveillance Frequency Control Program

4.0 DESIGN FEATURES

4.1 Site Location

The Palisades Nuclear Plant is located on property owned by Holtec Palisades, LLC on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor core shall contain 204 fuel assemblies. Each assembly shall consist of a matrix of zircaloy-4 or M5 clad fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide (UO_2) 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. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution. Poison may be placed in the fuel bundles for long-term reactivity control.

4.2.2 Control Rod Assemblies

The reactor core shall contain 45 control rods. Four of these control rods may consist of part-length absorbers. The control material shall be silver-indium-cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The Region I (See Figure B 3.7.16-1) Carborundum equipped fuel storage racks incorporating Regions 1A, 1B, 1C, 1D, and 1E are designed and shall be maintained with:

a. New or irradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.54 weight percent;

4.3.1 Criticality (continued)

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. Regions 1A, 1B, and 1C have a nominal 10.25 inch center to center distance between fuel assemblies;

e. Regions 1D and 1E have a nominal 11.25 inch by 10.69 inch center to center distance between fuel assemblies;

f. Region 1A is defined as a subregion of the Region I storage racks located in the main spent fuel pool and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1A shall be in a maximum of two-of-four checkerboard loading pattern of two fuel assemblies (or fissile bearing components) and two empty cells. Designated empty cells may contain non-fuel bearing components in accordance with Section 4.3.1.1m.2. below;

g. Region 1B is defined as a subregion of the Region I storage racks located in the main spent fuel pool and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1B shall be in a maximum of three-of-four loading pattern consisting of three fuel assemblies (or fissile bearing components) and one empty cell. Fuel assemblies in Region 1B shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-2. Designated empty cells may contain non-fuel bearing components in accordance with Section 4.3.1.1m.2. below;

h. Region 1C is defined as a subregion of the Region I storage racks located in the main spent fuel pool and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1C may be in a maximum of four-of-four loading pattern with no required empty cells. Fuel assemblies in Region 1C shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-3;

i. Interface requirements for the main spent fuel pool between Region 1A, 1B, and 1C are as follows. Region 1A, 1B, and 1C can be distributed in Region I, in the main spent fuel pool, in any manner provided that any two-by-two grouping of storage cells and the assemblies in them correspond to the requirements of 4.3.1.1f., 4.3.1.1g., or 4.3.1.1h. above;

4.3.1 Criticality (continued)

j. Region 1D is defined as a subregion of the Region I storage rack located in the north tilt pit and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1D may be in a maximum of three-of-four loading pattern consisting of three fuel assemblies (or fissile bearing components) and one empty cell. Fuel assemblies in Region 1D shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-4;

k. Region 1E is defined as a subregion of the Region I storage rack located in the north tilt pit and is subject to the following restrictions. Fuel assemblies (or fissile bearing components) located in Region 1E may be in a maximum of four-of-four loading pattern with no required empty cells. Fuel assemblies in Region 1E shall meet the enrichment dependent burnup restrictions listed in Table 3.7.16-5;

l. Interface requirements for the north tilt pit between Region 1D and 1E are as follows. Region 1D and 1E can be distributed in Region I in the north tilt pit in any manner provided that any two-by-two grouping of storage cells and the assemblies in them correspond to the requirements of 4.3.1.1j. or 4.3.1.1k. above;

m. Non-fissile bearing component restrictions are as follows:

1. Non-fissile material components may be stored in any designated fuel location in Region 1A, 1B, 1C, 1D, or 1E without restriction.

2. The following non-fuel bearing components (NFBC) may be stored face adjacent to fuel in any designated empty cell in Region 1A or 1B.

(i) The gauge dummy assembly and the lead dummy assembly may be stored face adjacent to fuel in any designated empty cells with no minimum required separation distance.

(ii) A component comprised primarily of stainless steel that displaces less than 30 square inches of water in any plane within the active fuel region may be stored in any designated empty cell as long as the NFBC is at least ten locations away from any other NFBC that is in a designated empty cell, with the exception of 4.3.1.1m.2.(i) above.

4.3.1 Criticality (continued)

3. Control blades may be stored in both fueled and unfueled locations in Regions 1D and 1E, with no limitation on the number.

4.3.1.2 The Region I (See Figure B 3.7.16-1) Metamic equipped fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. A nominal 10.25 inch center to center distance between fuel assemblies;

e. New or irradiated fuel assemblies;

f. Two empty rows of storage locations shall exist between the fuel assemblies in a Carborundum equipped rack and the fuel assemblies in an adjacent Metamic equipped rack; and

g. A minimum Metamic B^{10} areal density of 0.02944 g/cm^2.

4.3.1.3 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with:

a. Fuel assemblies having maximum nominal planar average U-235 enrichment of 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

d. A nominal 9.17 inch center to center distance between fuel assemblies; and

4.3 Fuel Storage

4.3.1 Criticality (continued)

e. New or irradiated fuel assemblies which meet the maximum nominal planar average U-235 enrichment, burnup, and decay time requirements of Table 3.7.16-1

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

a. Twenty four unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. Keff ≤ 0.95 when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

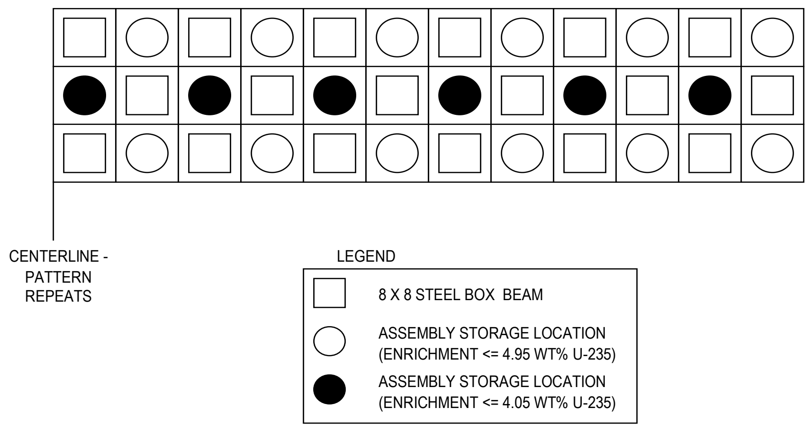
c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

4.3.2 Drainage

The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.



Note: If any assemblies containing fuel enrichments greater than 4.05% U-235 are stored in the New Fuel Storage Rack, the center row must remain empty.

Figure 4.3-1 (page 1 of 1)
New Fuel Storage Rack Arrangement

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage to the engineered safeguards rooms, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practical. The systems include the Containment Spray System, the Safety Injection System, the Shutdown Cooling System, and the containment sump suction piping. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

c. The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 255 psig.

d. Piping from valves CV-3029 and CV-3030 to the discharge of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig.

e. The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank.

5.5.3 (Deleted)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the Offsite Dose Calculation Manual (ODCM), (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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 ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.

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. Limitation on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each plant to unrestricted areas conforming to 10 CFR 50, Appendix I,

e. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

    1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

    2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

f. 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,

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each plant to areas beyond the site boundary conforming to 10 CFR 50, Appendix I,

h. Limitations on the annual doses 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 frequency.

5.5.5 Containment Structural Integrity Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Structural Integrity Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE and IWL.

If, as a result of a tendon inspection, corrective retensioning of five percent (8) or more of the total number of dome tendons is necessary to restore their liftoff forces to within the limits, a dome delamination inspection shall be performed within 90 days following such corrective retensioning. The results of this inspection shall be reported to the NRC in accordance with Specification 5.6.7, “Containment Structural Integrity Surveillance Report.”

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Containment Structural Integrity Surveillance Program inspection frequencies.

5.5.6 Primary Coolant Pump Flywheel Surveillance Program

a. Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the upper flywheels each 10 years.

b. The provisions of SR 3.0.2 are not applicable to the Flywheel Testing Program

5.5 Programs and Manuals

5.5.7 (Deleted)

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 of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged 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 SG 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.

5.5.8 Steam Generator (SG) Program

b. Performance criteria for SG tube integrity. (continued)

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 0.3 gpm.

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

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged. The following alternative repair criteria shall be applied as an alternate to the 40% depth based criteria:

1. Tubes found by inservice inspection to contain service induced flaws within 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service.

2. Tubes found by inservice inspection to contain service induced flaws within 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service.

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 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, and that may satisfy the applicable tube repair criteria. The tube to tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

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.

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 from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, 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. When the alternate repair criteria of TS 5.5.8c.1 are implemented, inspect 100% of the inservice tubes to the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8c.1 every 24 effective full power months, or one refueling outage, whichever is less.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and 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, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

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 (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Handling Area Ventilation (FHAV) systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 (RG 1.52), and in accordance with RG 1.52 and ASME N510-1989, at the system flowrates and tolerances specified below*:

a. Demonstrate for each of the ventilation systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
FHAV (single fan operation)	7300 ± 20%
FHAV (dual fan operation)	10,000 ± 20%
CRV	3,200 +10% -5%

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (Continued)

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989.

Ventilation System	Flowrate (CFM)
FHAV (dual fan operation)	10,000 ± 20%
CRV	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 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 and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
FHAV	6.00%	95%
CRV	0.157%	70%

d. For each of the ventilation systems, demonstrate 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 RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_20)	Flowrate (CFM)
FHAV (dual fan operation)	6.0	10,000 ± 20%
CRV	8.0	3200 +10% -5%

e. Demonstrate that the heaters for the CRV system dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
CRV	15 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

5.5 Programs and Manuals

5.5.11 Fuel Oil Testing Program

A 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 requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

    1. API gravity or an absolute specific gravity,

    2. Kinematic viscosity, and

    3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program.

5.5.12 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 FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5 Programs and Manuals

5.5.12 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.12.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).

5.5.13 Safety Functions 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. The SFDP shall contain the following:

a. 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;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

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:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (Continued)

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.

5.5.14 Containment Leak 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 NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, with the following exceptions:

   1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

      Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

   2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at ≥10 psig instead.

   3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 54.2 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

5.5 Programs and Manuals

5.5.14 Containment Leak Rate Testing Program (Continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage is $\leq 1.0\ L_a$ when tested at $\geq P_a$ and combined with all penetrations and valves subjected to Type B and C tests. However, during the first unit startup following testing performed in accordance with this program, the leakage rate acceptance criteria is $< 0.6\ L_a$ when combined with all penetrations and valves subjected to Type B and C tests.

b) For each Personnel Air Lock door, leakage is $\leq 0.023\ L_a$ when pressurized to ≥ 10 psig.

c) For each Emergency Escape Air Lock door, a seal contact check , consisting of a verification of continuous contact between the seals and the sealing surfaces, is acceptable.

e. "Containment OPERABILITY" is equivalent to "Containment Integrity" for the purposes of the testing requirements.

f. The provisions of SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

g. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5 Programs and Manuals

5.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program. This documentation shall contain:

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

b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant manager.

5.5 Programs and Manuals

5.5.16 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 Ventilation (CRV) Filtration, CRE 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 Sections 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. Measurement, 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 CRV Filtration, operating at the flow rate required by the Ventilation Filter Testing Program, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage 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 paragraph 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.

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 paragraphs c and d, respectively.

5.5 Programs and Manuals

5.5.17 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 (Deleted)

5.6.2 Radiological Environmental Operating Report

The Radiological Environmental Operating Report covering the operation of the plant during the previous calendar year shall be submitted before May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering operation of the plant in the previous year shall be submitted prior to May 1 of each year 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 plant. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual and Process Control Program, and shall be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 (Deleted)

5.6.5 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:

3.1.1	Shutdown Margin
3.1.6	Regulating Rod Group Position Limits
3.2.1	Linear Heat Rate Limits
3.2.2	Radial Peaking Factor Limits
3.2.4	ASI Limits
3.4.1	DNB Limits

5.6 Reporting Requirements

5.6.5 COLR (Continued)

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. EMF-96-029(P)(A) Volumes 1 and 2, “Reactor Analysis System for PWRs,” Siemens Power Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

2. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. (Bases report not approved) (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. EMF-84-093(P)(A), “Steam Line Break Methodology for PWRs, “Siemens Power Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

5. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company. (Bases document not approved) (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6. EMF-2310 (P)(A), Revision 0, Framatome ANP, Inc., May 2001, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors.” (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

7. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, & 3.2.2)

8. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

9. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.2.1, 3.2.2, & 3.2.4)

5.6 Reporting Requirements

5.6.5 COLR (Continued)

10. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

11. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

12. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWD/MTU," Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

13. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

14. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

15. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation. (LCOs 3.1.6, 3.2.1, & 3.2.2)

16. ANF-87-150 Volume 2, "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. [Approved for use in the Palisades design during the NRC review of license Amendment 118, November 15, 1988] (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.4.1)

17. EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors." (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

18. EMF-2328 (P)(A), Revision 0, Framatome ANP, Inc., March 2001, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." (LCOs 3.1.6, 3.2.1, & 3.2.2)

19. BAW-2489P, "Revised Fuel Assembly Growth Correlation for Palisades." (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

20. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." (LCOs 3.1.6, 3.2.1, & 3.2.2)

5.6 Reporting Requirements

5.6.5 COLR (Continued)

21. BAW-10240(P)-A, “Incorporation of M5 Properties in Framatome ANP Approved Methods.” (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

c. 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 Systems limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring 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 to OPERABLE status.

5.6.7 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 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 during the inspection outage for each active degradation mechanism,

5.6 Reporting Requirements

5.6.8 Steam Generator Tube Inspection Report (continued)

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

i. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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 or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

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, or

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

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)

4. 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, and 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 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

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 or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designee.

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.

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 or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

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, and with the means to communicate with and control every individual in the area, or

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)

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; and 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 options (2) and (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.

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.

PALISADES PLANT

RENEWED FACILITY OPERATING LICENSE DPR-20

APPENDIX B

ENVIRONMENTAL PROTECTION PLAN (NON-RADIOLOGICAL)

PALISADES PLANT

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

TABLE OF CONTENTS

Section Page

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during construction and operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the plant is operated in an environmentally acceptable manner, as established by the FES and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES which relate to water quality matters are regulated by way of the licensee's NPDES permit.

2.0 Environmental Protection Issues

In the final addendum to the FES-OL dated February 1978 the staff considered the environmental impacts associated with the operation of the Palisades Plant. Certain environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment.

2.1 Aquatic Issues

Specific aquatic issues raised by the staff in the FES-OL were:

The need for aquatic monitoring programs to confirm that thermal mixing occurs as predicted, that chlorine releases are controlled within those discharge concentrations evaluated, and that effects on aquatic biota and water quality due to plant operation are no greater than predicted.

Aquatic issues are addressed by the effluent limitations, and monitoring requirements are contained in the effective NPDES permit issued by the State of Michigan, Department of Natural Resources. The NRC will rely on this agency for regulation of matters involving water quality and aquatic biota.

2.2 Terrestrial Issues

1. Potential impacts on the terrestrial environment associated with drift from the mechanical draft cooling towers. (FES-OL addendum Section 6.3)

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such changes, tests or experiments do not involve an unreviewed environmental question, and do not involve a change in the Environmental Protection Plan. Changes in plant design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this section.

Before engaging in additional construction or operational activities which may affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation. When such activity involves a change in the Environmental Protection Plan, such activity and change to the Environmental Protection Plan may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental statement (FES) as modified by staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level [in accordance with 10 CFR Part 51.5(b)(2)] or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include a written evaluation which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question nor constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of his Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the NPDES Permits and State Certifications

Violations of the NPDES Permit or the State certification (pursuant to Section 401 of the Clean Mater Act) shall be reported to the NRC by submittal of copies of the reports required by the NPDES Permit or certification.

Changes and additions to the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The NRC shall be notified of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The notification of a licensee-initiated change shall include a copy of the requested revision submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, or local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and promptly reported to the NRC within 24 hours by telephone, telegraph, or facsimile transmissions followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Meteorological Monitoring

A meteorological monitoring program shall be conducted in the vicinity of the plant site for at least two years after conversion to cooling towers to document effects of cooling tower operation on meteorological variables. Data on the following meteorological variables shall be obtained from the station network shown in Figure 4.2.1: precipitation, temperature, humidity, solar radiation, downcoming radiation, visibility, wind direction and wind speed. In addition, studies shall be conducted for at least two years to measure affects of cooling tower drift on vegetation by associated salt deposition, icing or other causes.

5.0 Administrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the Environmental Protection Plan. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of plant operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to plant structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the plant. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Request for change in the Environmental Protection Plan shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the Environmental Protection Plan.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following issuance of the operating license. The period of the first report shall begin with the date of issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this Environmental Protection Plan for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends towards irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of action to alleviate the problem.

The Annual Environmental Operating Report shall also include:

(a) A list of EPP noncompliances and the corrective actions taken to remedy them.

(b) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental issue.

(c) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of nonroutine event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the time it is submitted to the other agency.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 276 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

HOLTEC PALISADES, LLC

PALISADES ENERGY, LLC

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated December 14, 2023 (Agencywide Documents Access and Management System Accession No. ML23348A148), as supplemented by letters dated July 9, 2024 (ML24191A422), and December 19, 2024 (ML24354A111), Holtec Decommissioning International, LLC (HDI), on behalf of Holtec Palisades, LLC[2] (collectively, Holtec), submitted a license amendment request to make certain changes to the Renewed Facility Operating License (RFOL) DPR-20 for the Palisades Nuclear Plant (PNP). Specifically, Holtec requested an amendment to revise the RFOL, the Permanently Defueled Technical Specifications (PDTS)[3], the Environmental Protection Plan (EPP), and the Physical Security Plan (PSP) to support resumption of power operations at PNP.

The supplements dated July 9, 2024 and December 19, 2024, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 7, 2024 (89 FR 64486).

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2 By letter dated July 24, 2025, the NRC issued Amendment No. 275, reflecting Palisades Energy, LLC, as the licensed operator. Holtec Palisades, LLC, remains the licensed owner of PNP.

3 Holtec submitted two separate license amendment requests (LARs) that would revise the PNP RFOL and PDTS. The LAR that is the subject of this safety evaluation (SE), in part, proposes to reinstate the applicable operational technical specification requirements (ML23348A148) into the PNP license previously in effect at PNP. The second LAR proposes to reinstate certain TS administrative controls, such as staffing and training requirements (ML24040A089), and is evaluated in a separate SE. Holtec elected to submit two LARs to retain a clear connection between these LARs and the previously-approved license amendments requests to transition the PNP license to a decommissioning status (ML17208A428 and ML21152A108, respectively).

1.1 Background Related to Holtec's Requests to Reauthorize Power Operations at Palisades

By letter dated January 4, 2017 (ML17004A062), pursuant to Paragraph (a)(1)(i) of Section 50.82, "Termination of license," of Title 10 of the Code of *Federal Regulations* (10 CFR), Entergy Nuclear Operations, Inc. (Entergy), the previous licensee for PNP, certified to the NRC that it decided to permanently cease power operations at PNP by October 1, 2018. By letters dated September 28, 2017 (ML17271A233), and October 19, 2017 (ML17292A032), Entergy updated its timeline and certified to the NRC that it planned to permanently cease power operations at PNP no later than May 31, 2022. By application dated December 23, 2020 (ML20358A075), as supplemented, Entergy on behalf of itself, Entergy Nuclear Palisades, LLC, Holtec International, and HDI submitted a license transfer application to transfer the PNP license from Entergy to Holtec. By letter dated December 13, 2021 (ML21292A145), the NRC issued an order consenting to the license transfer.

On May 20, 2022, PNP permanently ceased power operations. Pursuant to 10 CFR 50.82(a)(1)(ii), by letter dated June 13, 2022 (ML22164A067), Entergy certified to the NRC that all fuel had been permanently removed from the PNP reactor vessel and placed in the spent fuel pool (SFP) on June 10, 2022. These certifications were docketed by the NRC. Upon docketing the 10 CFR 50.82(a)(1) certifications, 10 CFR 50.82(a)(2) no longer authorizes operation of the reactor, or emplacement or retention of fuel into the reactor vessel. Shortly after PNP transitioned to a permanently shutdown and defueled facility in accordance with 10 CFR 50.82(a)(2), Holtec Palisades, LLC assumed ownership of PNP, and HDI became the licensed operator for decommissioning PNP (ML22173A173) and began the decommissioning process.

In early 2023, HDI engaged with the NRC staff regarding the potential restart of reactor operation at PNP. By letter dated March 13, 2023 (ML23072A404), HDI submitted its proposed regulatory path to resume power operations at PNP through a series of licensing and regulatory actions to restore the plant's licensing basis to the one in effect just prior to permanent shutdown.

Specifically, from September 2023 to May 2024, the NRC received the following licensing and regulatory requests related to the potential restart of Palisades:

- A September 28, 2023, request for an exemption "from the 10 CFR 50.82(a)(2) restriction that prohibits reactor power operations and retention of fuel in the reactor vessel … by allowing for a one-time rescission of the docketed 10 CFR 50.82(a)(1) certifications." (ML23271A140) (Exemption Request).

- A December 6, 2023, license transfer application, seeking NRC consent to, and a conforming amendment for, a transfer of operating authority from HDI to Palisades Energy, LLC under Renewed Facility Operating License No. DPR-20 for Palisades and the general license for the Palisades Independent Spent Fuel Storage Installation (ISFSI) (ML23340A161) (License Transfer Application).

- A December 14, 2023, license amendment request in support of resuming power operations that largely seeks to undo the changes made by the previously issued PDTS Amendment with some proposed differences from the previous operating reactor TS (ML23348A148) (Power Operations TS Amendment).

- A February 9, 2024, license amendment request in support of resuming power operations that largely seeks to undo the changes made by the previously issued Defueled Administrative Controls Amendment with some proposed differences from the previous operating reactor TS (ML24040A089) (Administrative Controls Amendment).

- A May 1, 2024, license amendment request to revise the Palisades site emergency plan to support resuming power operations (ML24122C666) (Emergency Plan Amendment).

- A May 24, 2024, license amendment request to revise the Palisades main steam line break analysis to "support the Palisades restart project." (ML24145A145) (MSLB Amendment).

While the changes requested in this LAR (Power Operations TS Amendment) are a necessary part of Holtec's regulatory approach to support reinstatement of the PNP power operations licensing basis, the NRC's approval of the proposed changes in this LAR is not sufficient to authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel. NRC approval of all of the licensing and regulatory requests listed above is necessary to reauthorize power operations at PNP. This SE addresses only the staff's review of the Power Operations TS Amendment. As discussed in Section 2.3 below, the other licensing and regulatory actions described above were reviewed separately by the NRC staff and are being issued concurrently with this amendment.

In February 2025, Holtec submitted two additional license amendment requests that Holtec states are necessary for the resumption of power operations at PNP. These amendments would revise certain technical specifications to support repairing of steam generator tubes by sleeving and revise PNP's licensing basis to incorporate a leak-before-break methodology (ML25043A348 and ML25035A216, respectively). In June 2025, Holtec submitted an additional amendment to revise the schedule in their license condition for full implementation of the NFPA-805 fire protection modifications (ML25175A275). These amendments are still currently under NRC review.

1.2 Proposed Changes to the Renewed Facility Operating License

In License Amendment No. 272, dated May 13, 2022 (ML22039A198), the NRC approved significant reductions to the technical specifications for PNP to reflect the reduced risk and lesser scope of operations associated with the permanently defueled status of the plant once it entered decommissioning. Amendment No. 272 became effective following the docketing of the 10 CFR 50.82(a)(1) certifications. The current LAR would effectively reverse the changes made by Amendment No. 272 and return the PNP RFOL, TS, and EPP to those in effect during plant operation, with certain exceptions described below.

In the LAR, Holtec proposed changes to reinstate all of the PNP License Conditions to those that were in effect prior to docketing the 10 CFR 50.82(a)(1) certifications for permanent cessation of operations and removal of fuel from the reactor vessel, with the following exceptions:

- License Condition 1.B is not reinstated because it is superseded.
- License Conditions 2.C.(4) and 2.C.(7) are not reinstated because they are considered historical License Conditions.

The proposed changes are evaluated in Section 3.3 of this safety evaluation.

1.3 Proposed Changes to the Permanently Defueled Technical Specifications

In the LAR, Holtec proposed changes to reinstate the PNP TS requirements to those that were in effect just prior to docketing 10 CFR 50.82(a)(1) certifications for permanent cessation of operations and removal of fuel from the reactor vessel, with the following exceptions:

- Appendix A Table of Contents is deleted.
- Surveillance Requirement 3.1.4.3 (Control Rod Movement) had an obsolete cycle specific note that is not reinstated.

The proposed changes are evaluated in Section 3.4 of this safety evaluation.

1.4 Proposed Changes to the Environmental Protection Plan

In the LAR, Holtec proposed changes to reinstate the PNP EPP to the version that was in effect just prior to docketing 10 CFR 50.82(a)(1) certifications for permanent cessation of operations and removal of fuel from the reactor vessel.

The proposed changes are evaluated in Section 3.5 of this safety evaluation.

1.5 Background and Proposed Changes to the Physical Security Plan

By letter dated September 14, 2022 (ML22257A097), HDI, on behalf of Holtec Palisades, LLC, requested an amendment to PNP RFOL No. DPR-20. The proposed license amendment would have revised the PNP RFOL to remove the Cybersecurity Plan (CSP) requirements contained in License Condition 2.E. HDI requested the change to support the decommissioning of PNP. By letter dated December 12, 2023 (ML23346A083), HDI withdrew the amendment request to revise RFOL No. DPR-20 to remove the CSP requirements contained in License Condition 2.E. The licensee withdrew the amendment request to support the resumption of power operations at PNP. The NRC staff acknowledged the request on December 26, 2023 (ML23355A124).

Required security plans at PNP include the Physical Security Plan, the Training and Qualification Plan (T&QP), the Safeguards Contingency Plan (SCP), and the CSP[4]. The "Palisades Nuclear Plant Physical Security Plan" includes the PSP, SCP, T&QP, and Independent Spent Fuel Storage Installation (ISFSI) Security Plan. The NRC staff collectively refers to the "Palisades Nuclear Plant Physical Security Plan" henceforth as the "PNPPSP."

The current LAR would reinstate PNPPSP Revision 16, dated June 11, 2014 (not publicly available; Safeguards Information (SGI)) at PNP. The LAR supplement dated July 9, 2024, modifies LAR Section 3.2.1, "Proposed Changes to the PNP Renewed Facility Operating License," page 8 of 97, as follows:

> Add as 4th paragraph: License Condition 2.E. for the PNP Physical Security Plan (PSP) is not proposed for modification by this power operations technical specification LAR and was not modified by the PDTS amendment. License Condition 2.E remains unchanged from the pre-permanent cessation of power

[4] This amendment request does not impact PNP's CSP; therefore, the NRC staff did not include the CSP for review as part of this safety evaluation. The NRC staff completed a full implementation cybersecurity inspection of PNP's CSP and installation in November 2020 (ML20325A356). Subsequently, the NRC staff placed PNP's program into the Reactor Oversight Process baseline inspection program. Palisades continued to maintain both the protected area and the Security Critical Digital Assets (CDAs) throughout decommissioning.

> operations / permanent removal of fuel from the PNP reactor power operations technical specifications version. However, the Palisades Physical Security Plan has been revised since certification of permanent defueling and permanent removal of fuel from the reactor to reflect the reduced risks of a reactor in decommissioning. To support the transition of PNP back to a power operations plant the PSP will be updated, in accordance with 10 CFR 50.54(p), *Conditions of licenses,* to reflect the docketed version that was in effect prior to the 10 CFR 50.82(a)(1) certifications, PSP Revision 16. Any PSP changes made during decommissioning that will be retained in the reinstated POLB PSP have been or will be evaluated in accordance with 10 CFR 50.54(p) against the PNP POLB to determine if NRC approval is required to retain the change prior to exiting the period of decommissioning. The power operations PSP revision will be implemented coincident with the implementation of the power operations technical specification (POTS) amendment.

The NRC staff understood the revised LAR to denote that PNP would update the existing decommissioning PNPPSP, Revision 21, in accordance with 10 CFR 50.54(p)(2) to align with PNPPSP, Revision 16. The update would retain or remove any modifications made to PNPPSP, Revision 16 during decommissioning using an evaluation process in accordance with 10 CFR 50.54(p) to determine if NRC approval is required to retain any changes prior to exiting the period of decommissioning. Subsequently, PNP would submit the revised version to NRC staff.

By letter dated March 26, 2025 (ML25087A023), HDI on behalf of Holtec Palisades LLC, submitted draft PNPPSP, Revision 22 to the NRC staff. By letter dated May 5, 2025 (ML25127A277), HDI on behalf of Holtec Palisades LLC, submitted an updated draft PNPPSP to replace the previously submitted draft. Revision 22 to the PNPPSP made changes to (1) the acronym used for Palisades from PLP to PNP, (2) the facility layout description, (3) the equipment used by security personnel, (4) definitions and acronyms used in the glossary of terms, (5) training requirements for security personnel, and (6) the Critical Task Matrix.

The requirement at 10 CFR 50.54(p)(2) does not mandate the licensee obtain approval from the NRC staff before making changes to a PSP, provided that these changes do not decrease the safeguards effectiveness of the plan. PNP submitted their revised PNPPSP to the NRC staff for review, not approval, prior to its implementation.

The proposed changes are addressed in Section 3.6 of this safety evaluation.

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance on which the NRC based its acceptance and evaluation of this LAR are contained in the following subsections.

2.1 Applicable Regulatory Requirements

Section 50.90, "Application for amendment of license, construction permit, or early site permit," of 10 CFR requires that whenever a licensee desires to amend the license, an application for an amendment must be filed with the Commission fully describing the changes desired and following as far as applicable, the form prescribed for original applications.

Under paragraph (a) of 10 CFR 50.92, Issuance of amendment," determinations on whether to grant a license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. Both the common standards for licenses and construction permits in paragraph (a) of 10 CFR 50.40, “Common standards,” (regarding, among other things, consideration of the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals), and those specifically for issuance of operating licenses in paragraph (a)(3) of 10 CFR 50.57, “Issuance of operating license,” provide that there must be “reasonable assurance” that the activities at issue will not endanger the health and safety of the public, and that the applicant will comply with the Commission’s regulations.

Section 50.36, “Technical specifications,” of 10 CFR establishes the regulatory requirements related to the content of TS. The five categories of items required to be in the TS are listed in 10 CFR 50.36(c) and include (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

Paragraph (b) of Section 50.36b, “Environmental conditions,” of 10 CFR states, in part, that an operating license may include conditions to protect the environment during operation and decommissioning. Obligations in the environmental area, include, as appropriate, requirements for reporting and keeping records of environmental data, and any conditions and monitoring requirement for the protection of the nonaquatic environment. These conditions are to be set out in an attachment to the permit or license, which is incorporated in and made a part of the permit or license. These conditions will be derived from information contained in the environmental report.

Regulations in 10 CFR 50.48, “Fire protection,” establish the fire protection requirements for operating reactors. As stated in 10 CFR 50.48(a): “Each holder of an operating license issued under this part…must have a fire protection plan that satisfies Criterion 3 of appendix A to this part.” Regulations in 10 CFR 50.48(c) provide the requirements for use of *National Fire Protection Association Standard NFPA 805* in creating a satisfactory fire protection plan. Section 3.3.8 of this SE evaluating License Condition 2.C.(3) addresses PNP’s compliance with these regulations.

Regulations in 10 CFR 50.51, “Continuation of license,” state, in part: “(a) [E]ach license will be issued for a fixed period of time to be specified in the license….” Section 3.3.18 of this SE evaluating the expiration date in the RFOL addresses PNP compliance with this regulation.

Regulations in 10 CFR 50.54(p)(1) state, “The licensee shall prepare and maintain safeguards contingency plan procedures in accordance with Appendix C of Part 73 of this chapter for affecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan. The licensee may not make changes which would decrease the effectiveness of a physical security plan, or guard training and qualification plan, or cyber security plan prepared under Section 50.34(c) or Section 52.79(a), or Part 73 of this chapter, or of the first four categories of information (Background, Generic Planning Base, Licensee Planning Base, Responsibility Matrix) contained in a licensee safeguards contingency plan prepared under § 50.34(d) or Section 52.79(a), or Part 73 of this chapter, as applicable, without prior approval of the Commission. A licensee desiring to make such a change shall submit an application for amendment to the license under Section 50.90.”

Regulations in 10 CFR 50.54(p)(2) state, in part, “The licensee may make changes to the plans referenced in paragraph (p)(1) of this section, without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of changes to the plans made without prior Commission approval for a period of 3 years from the date of the change, and shall submit, as specified in § 50.4 or § 52.3 of this chapter, a report containing a description of each change within 2 months after the change is made."

Regulations in 10 CFR Part 73, "Physical Protection of Plants and Materials," include performance-based and prescriptive regulatory requirements that, when adequately met and implemented, provide reasonable assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

Paragraph (c) of 10 CFR 50.34, "Contents of applications; technical information," 10 CFR 50.34(d), 10 CFR 50.54(p), and paragraph (c) of 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," require licensees to prepare and maintain a PSP, SCP, T&QP, and CSP. These plans describe the security-related actions the licensee will take to protect against acts of radiological sabotage.

Regulations in 10 CFR 72.212, "Conditions of general license issued under 72.210," establish the requirements for a general license, which is limited to that spent fuel which the general licensee is authorized to possess at the site under the specific license for the site.

Regulations in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission.

Regulations in 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," require that the design of the reactor vessel surveillance capsule program and withdrawal schedule must meet the requirements in the version of ASTM Standard Practice E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor pressure vessel (RPV) was purchased. Section 3.3.17 of this SE evaluating License Condition 2.J addresses PNP compliance with these requirements.

Regulations in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," contain leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. Section 3.3.13 of this SE evaluating License Condition 2.D addresses the manner in which PNP complies with this regulation.

2.2 Regulatory Guidance

The NRC staff's guidance for the review of operational TS is contained in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Chapter 16.0, "Technical Specifications," Revision 3, dated March 2010 (ML100351425). As described therein, as part of a regulatory standardization effort, the NRC staff has prepared standardized technical specifications (STS) for each of the LWR nuclear designs. Accordingly, the NRC staff's review of the current LAR includes consideration of whether the proposed changes for PNP are consistent with the applicable reference STS. The STS applicable to PNP is NUREG-1432, Revision 5.0, "Standard Technical Specifications, Combustion Engineering Plants," Volume 1, "Specifications," and Volume 2, "Bases," dated September 2021 (ML21258A421 and ML21258A424, respectively).

Regulatory guidance specific to the review of the PNPPSP includes:

- NUREG-0800, Chapter 13, Section 13.6.1, “Physical Security - Combined License and Operating Reactors,” Revision 2, dated August 2018 (ML17291B265).
- RG 5.7, Revision 1, “Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas,” dated May 1980 (ML003739976).
- RG 5.12, “General Use of Locks in the Protection and Control of Facilities and Special Nuclear Material,” dated October 1016 (ML15357A411).
- RG 5.44, Revision 3, “Perimeter Intrusion Alarm Systems,” dated October 1997 (ML003739217).
- RG 5.54, Revision 1, “Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants,” dated June 2009 (not publicly available; SGI).
- RG 5.62, Revision 3, “Reporting of Safeguards Events,” dated September 2024 (ML23299A176).
- RG 5.66, Revision 2, “Access Authorization Program for Nuclear Power Plants,” dated October 2011 (ML112060028).
- RG 5.69, “Guidance for the Application of the Radiological Sabotage Design-Basis Threat in the Design, Development, and Implementation of a Physical Security Protection Program that Meets 10 CFR 73.55 Requirements” (non-public; SGI).
- RG 5.71, “Cybersecurity Programs for Nuclear Power Reactors,” (ML22258A204).
- RG 5.74, “Managing the Safety/Security Interface,” dated April 2015 (ML14323A549).
- RG 5.75, “Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities,” (ML091690037).
- RG 5.76, “Physical Protection Programs at Nuclear Power Reactors,” dated July 2009 (not publicly available; SGI).
- RG 5.77, “Insider Mitigation Program,” Revision 1, dated September 2022 (ML16342B024).
- Nuclear Energy Institute (NEI) 03-12, Revision 7, “Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Installation Security Program” (not publicly available, SGI).
- NRC letter dated November 10, 2011, “Letter to D. Kline: Endorsement of Nuclear Energy Institute 03-12, “Template for Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [And Independent Spent Fuel Storage Installation Security Program]” Revision 7, October 2011” (ML112800383).

- NRC letter dated December 4, 2018, “NEI 03-12 Section 21 Revision 7.1 Approval for Use Letter” (ML18137A359).

2.3 NRC Staff’s Consideration of the Licensing and Regulatory Requests Related to the Reauthorization of Power Operations at PNP

The NRC staff’s consideration of all restart-related requests is governed by Commission-established policy on the reauthorization of reactor operations for plants in decommissioning. In denying a petition for rulemaking, Criteria To Return Retired Nuclear Power Reactors to Operations, (86 FR 24362; May 6, 2021), the Commission stated that “the NRC may consider requests from licensees to resume operations under the existing regulatory framework.” Further, the Commission stated that, “[i]f the NRC receives a request from the licensee for a decommissioning reactor to resume operations, the NRC would review the request consistent with applicable regulatory requirements. This review would include consideration of relevant safety standards to assure adequate protection of public health and safety.” In addition, in a decision related to the License Transfer Application, the Commission reaffirmed its policy that the NRC may consider licensee requests to resume operations under the existing regulatory framework (ML25119A109).

The NRC staff’s guidance for decommissioning nuclear power reactors in Regulatory Guide (RG) 1.184, “Decommissioning of Nuclear Power Reactors,” Revision 1 (ML13144A840), provides some limited guidance regarding potential restart requests. As described in Section C.2, “Certification of Permanent Cessation of Operations,” of RG 1.184, the NRC deems receipt of the certification of permanent cessation of operation as a commitment by the licensee to cease operations on the specified date. RG 1.184 states that following submission of the certification of permanent cessation of operations, or at any time during the decommissioning process, if the licensee desires to operate the facility again, the licensee must notify the NRC of its intentions in writing. It also states that the NRC would handle review and approval to return the facility to operation on a case-by-case basis, and the extent of review and approval would depend on the facility status at the time of the request to reauthorize operation.

Under the current requirements in 10 CFR 50.82, “Termination of license,” once a power reactor licensee has submitted written certifications to the NRC for both the permanent cessation of operations and the permanent removal of fuel from the reactor vessel, and the NRC has docketed those certifications, the 10 CFR Part 50 license no longer authorizes operation of the reactor. Following submission of the certifications of permanent cessation of operations and permanent removal of fuel from the reactor vessel, Holtec desired to operate the facility again and notified the NRC of its intentions in writing. Holtec is the first nuclear power plant licensee to notify the NRC of its intentions to request reauthorization of power operations after the filing and docketing of both 10 CFR 50.82(a)(1) certifications.

As discussed in Section 1.1 above, between September 2023 to May 2024, Holtec submitted an Exemption Request, a License Transfer Application, and four license amendment requests related to the potential restart of Palisades. The NRC staff has concluded, generally, that a facility licensee in decommissioning may apply to use the license amendment, transfer, and exemption processes, as applicable, to seek approval for the actions necessary to authorize the restart of a reactor in decommissioning. As discussed below, the staff has reached this conclusion for two principal reasons.

First, a facility operating license continues in effect for reactors in decommissioning because entering the decommissioning process involves a change in license authority and not a change to the form of the license itself. Specifically, 10 CFR. 50.82(a)(2) provides that upon docketing the § 50.82(a)(1) certifications, "the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel." In other words, the authority to operate is gone but the same Part 50 license remains. The continuation of the Part 50 license is made explicit by 10 CFR 50.51(b), which states "Each license for a facility that has permanently ceased operations, *continues in effect* beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the Commission notifies the licensee in writing that the license is terminated." Thus, the Palisades license remains a renewed Part 50 facility operating license during the decommissioning process.

Second, because the license for a reactor in decommissioning remains a facility operating license, licensing and regulatory requests within the existing regulatory framework may be used to restore the licensed authority for reactor operation. The license amendment, license transfer, and exemption processes are all within the existing regulatory framework and may be applied to a reactor in decommissioning as follows:

- Because license amendments are typically used to change the authorities and requirements for a reactor in decommissioning, the amendment process may be used to restore those authorities so long as the amendment standards in 10 CFR 50.92(a) are met.

- The license transfer process may be used to transfer authorities under the existing license to a transferee that is qualified to hold a license for reactor operation under 10 CFR 50.80(c)(1).

- Although 10 CFR 50.82(a)(2) prohibits reactor operation for a reactor in decommissioning, the exemption process established by 10 CFR 50.12 is available to remove regulatory restrictions, including the one in 10 CFR 50.82(a)(2), if all exemption requirements are met.

For these reasons, the NRC staff has concluded that a licensee in decommissioning may seek restart of reactor operation by applying to use relevant processes within the existing regulatory framework, including the license amendment, license transfer, and exemption processes.

Accordingly, separate from this SE, the NRC staff has reviewed and approved the Exemption Request (ML25163A182), the License Transfer Application (ML25167A245), the Administrative Controls TS Amendment (ML25157A107), the Emergency Plan Amendment (ML25150A281), and the MSLB Amendment (ML25156A045). The staff is issuing its approval of these actions concurrently with its approval of this LAR (Power Operations TS Amendment) to reauthorize power operations at PNP.

3.0 TECHNICAL EVALUATION

This SE section and associated subsections document the NRC staff's evaluation of Holtec's proposed changes in this LAR against the applicable regulations and guidance discussed in Section 2.0 of this SE.

In LAR Enclosure Section 2.1, "Reason for Proposed Change," Holtec explained that this LAR is needed to reinstate the PNP RFOL, TS, and EPP that were in effect just prior to the docketing of the 10 CFR 50.82(a)(1) certifications to support returning PNP to a power operations licensing basis. In addition, in LAR Enclosure Section 2.1, Holtec explained that no major decommissioning activities have occurred at PNP during this period when power operations and fuel placement in the reactor were prohibited.

According to Holtec, the shutdown of PNP was done for business and economic reasons and not because of the licensee's safety concerns (see Section 7.6 of ML23271A140). PNP was operating safely prior to its shutdown for decommissioning. The most recent operations phase PNP Annual Assessment Letter dated March 2, 2022 (ML22055B137), provides in part, the following assessment results:

> The NRC concluded that overall performance at your facility preserved public health and safety. The baseline inspection program was completed at your facility as defined in Inspection Manual Chapter 2515, "Light-Water Reactor Inspection Program - Operations Phase."
>
> The NRC determined the performance at Palisades Nuclear Plant, during the most recent quarter was within the Licensee Response Column, the highest performance category of the NRC's Reactor Oversight Process (ROP) Action Matrix, because all inspection findings had very low safety significance (i.e., Green), and all PIs [performance indicators] were within the expected range (i.e., Green).

Informed by the discussion above, the NRC staff's review approach focused on evaluating whether Holtec's proposed changes seeking to reinstate PNP's RFOL license conditions, TS, and EPP requirements that were in effect just prior to the docketing of the 10 CFR 50.82(a)(1) certifications, with some exceptions, are consistent with the previous NRC-approved PNP power operation RFOL, TS, and EPP, and are acceptable to support power operations at PNP.

3.1 Updated Final Safety Analysis Report for Resumption of Power Operations

The PNP design and licensing basis, including its relationship to the General Design Criteria are described in the Updated Final Safety Analysis Report (UFSAR), Revision 35 and other plant-specific licensing basis documents. Holtec states that PNP UFSAR, Revision 35, will be reinstated. This will include reinstatement of accident analyses and the safety reclassification of structures, systems, and components required to support the PNP power operations licensing basis. Changes made to the UFSAR after Revision 35 will be evaluated for retention, to the extent appropriate for an operating plant.

Specifically, in LAR Enclosure Section 2.2, "Description of Proposed Change," Holtec explained that the Updated Final Safety Analysis Report, currently titled the Defueled Safety Analysis Report (DSAR), will be updated, via the 10 CFR 50.59, "Changes, tests and experiments," process to reflect the docketed version that was in effect just prior to the docketing of the 10 CFR 50.82(a)(1) certifications, which is PNP UFSAR, Revision 35 (package, ML21125A344). Updating the PNP safety analysis report to reflect PNP in a power operating condition is necessary to support returning PNP to a power operations licensing basis. PNP UFSAR Revision 35 contains the analyses and evaluations from which the PNP power operation TS requested in the current LAR were derived. Thus, the NRC staff finds that reinstating the PNP UFSAR to the version that was in effect just prior to commencing the recent decommissioning process would ensure that the PNP power operations TS requested in this LAR are derived from the analyses and evaluation included in a safety analysis report that supports PNP power operations in accordance with 10 CFR 50.34(b).

During its review, the NRC staff noted that Section 2.2 further states:

> Any DSAR retained changes to UFSAR Revision 35 have been or will be evaluated via the 10 CFR 50.59 process against UFSAR Revision 35 to

> determine if NRC approval is required prior to exiting the period of decommissioning. This will include reinstatement of accident analyses and the safety classification of systems, structures, and components (SSCs), required to support the PNP power operations licensing basis (POLB). Changes made to the UFSAR after Revision 35 will be evaluated for retention, to the extent appropriate for an operating plant. The DSAR change back to the PNP POLB UFSAR will be accomplished under the 10 CFR 50.59 process and be implemented coincident with the associated license amendments.

This statement makes clear that Holtec intends to make or retain other UFSAR changes in addition to restoring the UFSAR to Revision 35, but the LAR does not specifically describe these changes or explain why they are acceptable.

By letter dated November 22, 2024 (ML24358A148), the NRC staff requested additional information regarding the specific changes that would be retained from the DSAR into the reconstituted version of PNP UFSAR, Revision 35. Holtec responded in a letter dated December 19, 2024 (ML24354A111), which (1) specifically described any planned differences from the content of UFSAR, Revision 35 (including the planned retention of DSAR changes to UFSAR, Revision 35) and (2) explained why these differences from Revision 35 are acceptable and satisfy the standards in 10 CFR 50.92(a) for license amendments. This supplemental letter also identified and described ongoing, planned, and pending activities that are anticipated to result in updates to the pre-shutdown PNP UFSAR, Revision 35, which will be evaluated under 10 CFR 50.59 and incorporated into the PNP UFSAR upon implementation of the PNP power operations TS discussed in this SE.

The NRC staff reviewed the specific changes that will be retained between the DSAR and PNP UFSAR, Revision 35, as well as those additional changes incorporated into the UFSAR to support the transition to power operations, and determined that all of the retained changes fall into one of the following categories: (1) updates to chronological licensing and site history (e.g., transfer of ownership of PNP from Entergy Nuclear Palisades, LLC to Holtec Palisades, LLC.); (2) retention of editorial changes and other administrative changes related to the retitling of historic information; (3) corrections or additions to plant information that have changed since Revision 35 (e.g., transmission system ownership); (4) updates to several radiation protection instrumentation discussions to increase accuracy; and (5) changes to the site organization structure to align with an operational stance. Based on its review, the NRC staff finds that the proposed changes to PNP UFSAR, Revision 35, including the DSAR retentions, are administrative in nature, and therefore, would have no impact on the analyses and evaluation included in UFSAR, Revision 35, from which the power operations TS requested in the LAR are derived. Based on these findings, the NRC staff concludes that Holtec's proposal to reinstate UFSAR, Revision 35, along with the proposed changes described above, to support power operations at PNP meets the requirements of 10 CFR 50.34(b). Therefore, the NRC staff finds the proposed changes acceptable.

The PNP UFSAR is a licensee-controlled document that is subject to the change control provisions of 10 CFR 50.59. Under 10 CFR 50.59, licensees are allowed to make changes in the facility and procedures as described in the final safety analysis report (FSAR, as updated) and conduct tests or experiments not described in the FSAR (as updated), without obtaining a license amendment pursuant to 10 CFR 50.90 provided specific criteria are met. As such, any future updates to the PNP UFSAR that require license amendments will be reviewed as separate licensing activities and are not reviewed in this safety evaluation.

3.2 Accident and Transient Analyses in the PNP UFSAR

As noted in Section 3.1 of this SE, Holtec proposed that the PNP UFSAR will be reinstated to Revision 35, which is the last docketed version of the UFSAR that was in effect prior to the 10 CFR 50.82(a)(1) certifications being docketed, along with certain administrative changes. In LAR Enclosure Section 3.1, "Accident and Transient Analyses Applicable to the Proposed Change," Holtec states, "[t]his includes reinstatement of the DBA [design-basis accident] and transient scenarios in Chapter 14. These scenarios and analyses are reinstated as they existed in Revision 35 of the UFSAR. No changes to these analyses are made as part of the return to power for PNP." In LAR Enclosure Section 3.1, Table 3-1, "PNP DBAs and Transients," Holtec listed the PNP UFSAR, Revision 35, Chapter 14 safety analysis DBAs and transients. The majority of these analyses were not changed during the transition to decommissioning; only portions of two analyses were revised to reflect the permanently defueled status of PNP (fuel assembly drop in the spent fuel pool and waste gas decay tank failure). These two analyses were changed because, during decommissioning, the fuel assembly drop and waste gas decay tank failure scenarios become the only remaining postulated accidents analyzed in the UFSAR and are the basis for several of the changes made in the transition to the permanently defueled technical specifications. The reinstatement of the power operations technical specifications will negate these decommissioning-specific changes and return all the analyses to those contained in the PNP UFSAR, Revision 35.

The LAR further states that the analyses for these scenarios demonstrate that the PNP plant design supports safe power operations and retention of fuel in the reactor vessel, and that radiological consequences from the postulated accident scenarios do not exceed the regulatory requirements of 10 CFR 50.67, "Accident source term," or 10 CFR Part 100, "Reactor Site Criteria," as applicable. PNP UFSAR, Revision 35, Chapter 14 contains the DBA and transient scenarios applicable to PNP during plant operations. It also describes the analyses that were performed to demonstrate that the plant could be operated safely and that radiological consequences from postulated DBAs do not exceed the applicable limits. Certain SSCs are credited in these analyses for the purpose of mitigating the DBAs or transients. These SSCs are considered for inclusion in TS per the requirements of 10 CFR 50.36(c) as discussed in Section 3.4 of this SE. Holtec also notes that as a normal part of evaluating core designs and operational parameters, reviews of Chapter 14 sections are performed for each fuel reload cycle by the nuclear fuel vendor to determine if the proposed core design is still bounding or if it needs to be reanalyzed; the UFSAR is updated accordingly, as needed.

The NRC staff reviewed the information provided in LAR Enclosure Section 3.1. The NRC staff finds that the safety analyses (e.g., DBAs and transients) proposed for reinstatement are the same as those previously described in UFSAR, Revision 35. Thus, when the PNP UFSAR, Revision 35 is reinstated, the safety analyses report will reflect the licensing basis as it existed just prior to the docketing of the 10 CFR 50.82(a)(1) certifications to support returning PNP to a power operation licensing basis. The staff also confirmed that the accident and transient analyses and evaluation contained in the safety analysis report formed the basis for the applicable TSs that existed prior to the defueled period and will now form the basis for the TSs proposed in this LAR (the staff evaluated the TS separately in Section 3.4 of this safety evaluation). Therefore, the NRC staff concludes that the accident and transient analyses in Chapter 14 of the PNP UFSAR, Revision 35, proposed for reinstatement, are acceptable.

3.3 Proposed Changes to the Renewed Facility Operating License

In Section 2.2 of the LAR Enclosure, Holtec stated that the proposed changes would revise certain requirements contained within the PNP RFOL to reinstate requirements that are necessary for power operation and revise or remove requirements that would no longer be applicable. Thus, the NRC staff's approach for its review of the proposed changes to the PNP RFOL focused on evaluating whether Holtec's proposed changes would appropriately reinstate the PNP RFOL license conditions, with some exceptions, to those PNP RFOL license conditions that were in effect just prior to the docketing of the 10 CFR 50.82(a)(1) certifications, such that they are consistent with the previously NRC-approved PNP power operations RFOL. To facilitate this review approach, the NRC staff compared the PNP RFOL license conditions proposed to be reinstated (with some exceptions) for consistency with the previous NRC-approved PNP RFOL license conditions that were removed by issuance of PNP License Amendment No. 272. Amendment No. 272 became effective following the docketing of the 10 CFR 50.82(a)(1) certifications.

Holtec's proposed changes to the PNP RFOL are described in LAR Enclosure Section 3.2.1, "Proposed Changes to the PNP Renewed Facility Operating License," and shown in attachments to the Enclosure: Attachment 1 (mark-up pages) and Attachment 2 (retyped pages). The LAR Enclosure Section 3.2.1 is arranged in a table format that identifies the current RFOL license condition, the proposed license condition, and provides a basis for the proposed change. Holtec also submitted supplemental information to provide further clarity and administrative corrections to the initial LAR documentation. The NRC staff's review of Holtec's proposed changes to the PNP RFOL license conditions is provided in the following Section 3.3 subsections of this SE.

As previously discussed, Holtec submitted a separate application to the NRC for the transfer of the PNP operating authority from HDI to a new entity, Palisades Energy, LLC. On July 24, 2025, the NRC staff issued an order approving the transfer (ML25167A243). Holtec Palisades, LLC remains the licensed owner of PNP. In the staff's evaluation of the proposed license conditions discussed below, the references to "HDI" in the RFOL license conditions are replaced by bracketed Palisades Energy, LLC, or Palisades Energy (e.g. [Palisades Energy]) to reflect the change in operating authority from HDI to Palisades Energy, as approved by the NRC staff's order approving the license transfer, and as reflected in Amendment No. 275 to the RFOL, the conforming amendment to the license transfer. The NRC staff's review of the change in operating authority is documented in a separate safety evaluation reviewing the license transfer application (ML25167A268), and therefore, these changes are not reviewed in this SE.

3.3.1 License Condition 1.B

Section 3.2.1 of the LAR Enclosure describes the proposed changes for RFOL License Condition 1.B. In this LAR enclosure section, Holtec explained that this license condition is currently deleted and will not be reinstated to what was in effect prior to the 10 CFR 50.82(a)(1) certifications being docketed because it has been superseded. By letter dated February 21, 1991 (ML020810482), the Commission issued Facility Operating License No. DPR-20 to the licensee for operation of PNP, which superseded Provisional Construction Permit No. CPPR-25. Accordingly, the NRC staff finds it acceptable to not reinstate PNP License Condition 1.B.

3.3.2 License Condition 2.B.(1)

Currently, License Condition 2.B.(1) reads:

> Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec Palisades to possess and use, and (b) HDI to possess and use the facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

Proposed License Condition 2.B.(1) would read as follows:

> Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec Palisades to possess and use, and (b) [Palisades Energy] to possess, use and operate, the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that reinstating this license condition would enable PNP to resume power operation as a utilization facility as defined by 10 CFR Part 50.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.B.(1) are acceptable.

3.3.3 License Condition 2.B.(2)

Currently, License Condition 2.B.(2) reads:

> HDI, pursuant to the Act and 10 CFR Parts 40 and 70, to possess source and special nuclear material that was used as reactor fuel, in accordance with the limitations for storage as described in the Updated Final Safety Analysis Report, as supplemented and amended;

Proposed License Condition 2.B.(2) would read as follows:

> [Palisades Energy], pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that reinstating this license condition would permit PNP to receive and use special nuclear material (SNM) as reactor fuel for reactor operation, and allow possession of SNM as reactor fuel in the amount required for reactor operation.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.B.(2) are acceptable.

3.3.4 License Condition 2.B.(3)

Currently, License Condition 2.B.(3) reads:

> HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources that were used for reactor startup, sealed sources that were used for reactor instrumentation and are used in the calibration of radiation monitoring equipment, and that were used as fission detectors in amounts as required;

Proposed License Condition 2.B.(3) would read as follows:

> [Palisades Energy] pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that reinstating this license condition would authorize the receipt and use of byproduct, source, and SNM as sealed neutron sources for activities such as reactor startup, reactor instrumentation, and fission detectors in support of power operation.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.B.(3) are acceptable.

3.3.5 License Condition 2.B.(5)

Currently, License Condition 2.B.(5) reads:

> HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operations of the facility.

Proposed License Condition 2.B.(5) would read as follows:

> [Palisades Energy] pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that reinstating this license condition reflects that PNP will become a power operation plant and produce byproduct and special nuclear materials in the course of plant operation.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.B.(5) are acceptable.

3.3.6 License Condition 2.C.(1)

Currently, License Condition 2.C.(1) reads:

> [deleted]

Proposed License Condition 2.C.(1) would read as follows:

> [Palisades Energy] is authorized to operate the facility at steady state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that reinstating this license condition reflects the operating condition and technical specifications for an operating plant, which will not be changed as part of the reauthorization of power operations at PNP. The license condition on maximum power level is consistent with that used to support the accident analyses evaluated in the UFSAR.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.C.(1) are acceptable.

3.3.7 License Condition 2.C.(2)

Currently, License Condition 2.C.(2) reads:

> The Technical Specifications contained in Appendix A, as revised through Amendment No. 273, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. HDI shall maintain the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Proposed License Condition 2.C.(2) would read as follows:

> The Technical Specifications contained in Appendix A, as revised through Amendment No. XXX, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. [Palisades Energy] shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that the current amendment number, No. 273, is replaced with "XXX," which acts as a placeholder for the new amendment number associated with approval of this LAR.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.C.(2) are acceptable.

3.3.8 License Condition 2.C.(3)

Currently, License Condition 2.C.(3) reads:

> [deleted]

Proposed License Condition 2.C.(3) would read as follows:

> <u>Fire Protection</u>
>
> [Palisades Energy] shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment requests dated December 12, 2012, November 1, 2017, November 1, 2018, and March 8, 2019, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014, August 14, 2014, November 4, 2014, December 18, 2014, and January 24, 2018, and May 28, 2019, as approved in the safety evaluations dated February 27, 2015, February 27, 2018, and August 20, 2019. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy

the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1x10-7/year (yr) for CDF and less than 1x10-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

   The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- “Fire Alarm and Detection Systems” (Section 3.8);
- “Automatic and Manual Water-Based Fire Suppression Systems” (Section 3.9);
- “Gaseous Fire Suppression Systems” (Section 3.10); and
- “Passive Fire Protection Features” (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee’s fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated February 27, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) <u>Transition License Conditions</u>

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2, below, risk-informed changes to the licensee’s fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-2, “Plant Modifications Committed,” of Entergy Nuclear Operations, Inc. (ENO) letter PNP 2019-028 dated May 28, 2019, to complete the transition to full compliance with 10 CFR 50.48(c) before the end of the refueling outage following the fourth full operating cycle after NRC approval. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Table S-3, “Implementation Items,” of ENO letter PNP 2014-097 dated November 4, 2014, within six months after NRC approval, or six months after a refueling outage if in progress at the time of approval with the exception of Implementation Items 3 and 8 which will be completed once the related modifications are installed and validated in the PRA model.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that this license condition is reinstated to provide requirements for implementation of a fire protection program that complies with paragraphs (a) and (c) of 10 CFR 50.48, "Fire protection," including requirements for risk informed changes that may be made without prior NRC approval, other changes that may be made without prior NRC approval, and the reinstatement of transition license conditions for implementation of NFPA 805. Holtec also stated that there is an administrative revision to the license condition to spell out the first occurrence of ENO as Entergy Nuclear Operations, Inc. when referring to NRC letters previously issued to ENO when it was the license holder for PNP. Holtec further stated that this license condition, which is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shut down of the reactor in the event of a fire, will become applicable at PNP with the resumption of power operation.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the NRC staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, except for the administrative change of spelling out ENO, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.C.(3) are acceptable.

3.3.9 License Condition 2.C.(4)

In the introductory paragraph of Section 3.2.1 of the LAR Enclosure, Holtec explained that License Condition 2.C.(4) was removed in Amendment No. 272 because it was identified as a historical license condition (a requirement that had expired). Holtec stated that this condition will remain identified as "deleted" and will not be reinstated to what was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications because it has expired.

Based on its review, the NRC staff concludes that not restoring the license condition to what was in effect at PNP just prior to the issuance of Amendment No. 272 is acceptable because the License Condition 2.C.(4) requirement has expired. Therefore, the NRC staff finds that retaining the current "deleted" designation of License Condition 2.C.(4) is acceptable.

3.3.10 License Condition 2.C.(5)

Currently, License Condition 2.C.(5) reads:

> Movement of a fuel cask in or over the spent fuel pool is prohibited when irradiated fuel assemblies decayed less than 90 days are in the spent fuel pool.

Proposed License Condition 2.C.(5) would read as follows:

> [deleted]

In Section 3.2.1 of the LAR Enclosure, Holtec explained that this license condition was added (see Amendment No. 272) to prohibit movement of a fuel cask in or over the spent fuel pool when irradiated fuel assemblies with less than 90 days decay time are in the spent fuel pool (SFP). Previously, TS 3.7.10, "Control Room Ventilation (CRV) Filtration," and TS 3.7.11, “Control Room Ventilation (CRV) Cooling,” were required during movement of a fuel cask in or over the SFP; TS 3.7.12, “Fuel Handling Area Ventilation System,” was required during movement of a fuel cask in or over the SFP when fuel assemblies with less than 90 days decay time are in the fuel handling building. These TS were deleted in Amendment No. 272 as part of the transition to decommissioning; therefore, this license condition was needed to control the timing of cask movement in or over the spent fuel pool. TS 3.7.10, TS 3.7.11, and TS 3.7.12 are being reinstated as part of this LAR as they are part of the PNP power operation TS (see SE Section 3.4.13). These TS will contain the limits on cask movement in or over the spent fuel pool. Therefore, Holtec stated that this decommissioning license condition is no longer needed to control cask movement. Holtec further stated that reinstatement of these TS will ensure an appropriate spent fuel assembly decay time requirement is maintained, and the associated analyses presented in UFSAR, Revision 35, Table 14.1-6 remain bounding.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the NRC staff finds, given that TS 3.7.10, TS 3.7.11, and TS 3.7.12 are proposed to be reinstated, that the retention of License Condition 2.C.(5), which was implemented by Amendment No. 272, is not necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed deletion of License Condition 2.C.(5) is acceptable.

3.3.11 License Condition 2.C.(7)

In the introductory paragraph of Section 3.2.1 of the LAR Enclosure, Holtec explained that License Condition 2.C.(7) was removed in Amendment No. 272 because it was identified as a historical license condition (a requirement that has been met). Holtec stated that License Condition 2.C.(7) will remain identified as “deleted” and will not be reinstated to what was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications because it has been met in accordance with the schedule specified in the license condition.

Based on its review, the NRC staff concludes that not restoring the license condition to what was in effect at PNP just prior to the issuance of Amendment No. 272 is acceptable because the License Condition 2.C.(7) requirement has been met in accordance with the schedule specified in the license condition. As such, this license condition no longer serves a regulatory purpose. Therefore, the NRC staff finds that retaining the current “deleted” designation of License Condition 2.C.(7) is acceptable.

3.3.12 License Condition 2.C.(8)

Currently, License Condition 2.C.(8) reads:

> [deleted]

Proposed License Condition 2.C.(8) would read as follows:

> Amendment 257 authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications in order to restore the PNP power operations RFOL. In addition, Holtec explained that this license condition is reinstated in its entirety to reflect the regulatory requirements for an operating plant because 10 CFR 50.61a, “Alternate fracture toughness requirements for protection against pressurized thermal shock events,” applies to pressurized water reactors for which an operating license has been issued.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.C.(8) are acceptable.

3.3.13 License Condition 2.D

Currently, License Condition 2.D reads:

> [deleted]

Proposed License Condition 2.D would read as follows:

> The facility has been granted certain exemptions from Appendix J to 10 CFR Part 50, “Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors.” This section contains leakage test requirements, scheduled and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted in a letter dated December 6, 1989.
>
> These exemptions granted pursuant to 10 CFR 50.12, are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that this license condition is reinstated in its entirety to reflect the operating condition of the facility because during power operation the containment plays a role in mitigating the consequences of the DBAs discussed in the PNP UFSAR, Revision 35. Appendix J to 10 CFR Part 50 provides leakage test requirements for the containment. Holtec stated that the exemptions previously approved by the NRC for use at PNP related to Appendix J are provided in this license condition, and that this license condition is necessary for PNP to resume power operations.

Amendment No. 272 deleted License Condition 2.D that referenced the exemptions to Appendix J that were previously granted by the NRC by letter dated December 6, 1989 (ML020810486). However, the exemptions were not formally rescinded. The NRC staff reviewed the exemptions and determined that the underlying bases supporting the exemptions are applicable to PNP in an operating status. Therefore, the staff finds that reinstatement of the license condition referencing these exemptions is acceptable.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR. Based on its review, the staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.D are acceptable.

3.3.14 License Condition 2.E

Currently, License Condition 2.E reads:

> HDI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revision to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21 [" Protection of Safeguards Information: Performance Requirements"], is entitled: "Palisades Nuclear Plant Physical Security Plan."
>
> HDI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Palisades CSP was approved by License Amendment No. 243 as supplemented by changes approved by License Amendment Nos. 248, 253, 259, and 264.

Proposed License Condition 2.E would read as follows:

> [Palisades Energy] shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revision to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21 [" Protection of Safeguards Information: Performance Requirements"], is entitled: "Palisades Nuclear Plant Physical Security Plan."
>
> [Palisades Energy] shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Palisades CSP was approved by License Amendment No. 243 as supplemented by changes approved by License Amendment Nos. 248, 253, 259, and 264.

Holtec explained that this license condition is not proposed for modification as part of this LAR and was not modified by Amendment No. 272. However, the PNPPSP has been revised since docketing the 10 CFR 50.82(a)(1) certifications, and as described in Section 1.5 and evaluated in Section 3.6 of this SE, the PNPPSP will be updated in accordance with 10 CFR 50.54(p) to reflect the docketed version of this plan that was in effect prior to entering decommissioning, which was PSP Revision 16. Holtec states that PSP changes made during decommissioning that will be retained in the reinstated power operations PSP have been evaluated in accordance with 10 CFR 50.54(p) against the PNP power operations licensing basis to determine if NRC approval is required to retain the change. The power operations PSP revision will be implemented coincident with the implementation of the Power Operations TS Amendment.

The NRC staff reviewed Holtec's proposal related to this license condition, as described above and shown in the supplement to the LAR dated July 9, 2024. Based on its review, the NRC staff finds that retaining this license condition is necessary to support the resumption of power operations at PNP to ensure that the requirements for the physical security, training and qualification, safeguards contingency, and cyber security plans are met. Therefore, the NRC staff concludes that the proposed change to License Condition 2.E is acceptable.

3.3.15 License Condition 2.H

Currently, License Condition 2.H reads:

> [deleted]

Proposed License Condition 2.H would read as follows:

> The Updated Safety Analysis Report supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the Updated Safety Analysis Report required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, [Palisades Energy] may make changes to the programs and activities described in the supplement without prior Commission approval, provided that [Palisades Energy] evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

In a supplement dated July 9, 2024 (ML24191A422, Item 5) to the initial LAR, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that reinstatement of this license condition supports plans to pursue subsequent license renewal (not evaluated as part of this SE) and is consistent with subsequent license renewal industry precedent.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR, as supplemented. Based on its review, the NRC staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.H are acceptable.

3.3.16 License Condition 2.I

Currently, License Condition 2.I reads:

> [deleted]

Proposed License Condition 2.I would read as follows:

> The Updated Safety Analysis Report supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. [Palisades Energy] shall complete these activities no later than March 24, 2011, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

In a supplement dated July 9, 2024 (ML24191A422, Item 5) to the initial LAR, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that reinstatement of this license condition supports plans to pursue subsequent license renewal (not evaluated as part of this SE) and is consistent with subsequent license renewal industry precedent.

The NRC staff reviewed the proposed changes to this license condition, as described above and shown in the attachments to the LAR, as supplemented. Based on its review, the NRC staff finds that the proposed changes are consistent with the previous NRC-approved license condition in the power operation RFOL that was in effect at PNP just prior to the issuance of Amendment No. 272. Therefore, the NRC staff concludes that the proposed changes to License Condition 2.I are acceptable.

3.3.17 License Condition 2.J

Currently, License Condition 2.J reads:

> [deleted]

Proposed License Condition 2.J would read as follows:

> All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, Holtec explained that this license condition is reinstated to address the requirements of 10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”

The NRC staff reviewed Holtec’s proposed change to this license condition, as described above and shown in the attachments to the LAR. This license condition was originally issued concurrent with the PNP RFOL on January 17, 2007. This license condition is described in Section 1.7, “Summary of Proposed License Conditions,” of NUREG-1871, “Safety Evaluation Report Related to the License Renewal of Palisades Nuclear Plant,” which was issued in January 2007 (ML070600578).

The regulations in 10 CFR Part 50, Appendix H, require that the design of the reactor vessel surveillance capsule program and withdrawal schedules meet the requirements in the version of ASTM Standard Practice E 185 that is current on the issue date of the ASME Code to which the RPV was purchased. The rule also requires the licensee to perform capsule testing and to report the test results in accordance with the requirements of ASTM E 185-82 to the extent practicable for the configuration of the test specimen in the RPV surveillance capsules.

The requirements in 10 CFR Part 50, Appendix H, are only relevant to nuclear power plants that are authorized to operate in the reactor-critical operating mode because that is the plant operating mode that produces high energy neutrons as a result of the reactor's nuclear fission process, and the requirements are set in place to provide assurance that the RPV will maintain adequate levels of fracture toughness throughout the operating life of the reactor. This license condition was imposed with the assumption that PNP would be operating for an additional 20 years under the RFOL (i.e., to and inclusive of March 24, 2031).

Based on its review of the proposed reinstatement of License Condition 2.J, the NRC staff concludes that implementation of the applicable surveillance capsule testing and reporting requirements is necessary for PNP to support power operation. Therefore, the NRC staff finds the addition of License Condition 2.J acceptable.

3.3.18 License Condition 2.K

Currently, License Condition 2.K reads:

> This license is effective as of the date of issuance and until the Commission notifies the licensee in writing that the license is terminated.

Proposed License Condition 2.J would read as follows:

> This license is effective as of the date of issuance and shall expire at midnight March 24, 2031.

In Section 3.2.1 of the LAR Enclosure, Holtec proposed to reinstate this license condition in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications to restore the PNP power operations RFOL. In addition, the Holtec explained that reinstatement of this license condition would comply with 10 CFR 50.51, "Continuation of license," which requires an operating reactor license to be issued for a fixed period of time. As such, Holtec concludes this change is necessary to resume power operations.

The NRC staff reviewed the proposed change to License Condition 2.K. The current License Condition 2.K, which addresses a permanently shut down and defueled condition of the facility in the process of decommissioning, is no longer necessary for a facility licensed to operate. The revised License Condition 2.K documents the date of expiration of the RFOL, consistent with requirements in 10 CFR 50.51. Therefore, the NRC staff finds the proposed change to License Condition 2.K acceptable.

3.4 Proposed Changes to the Permanently Defueled Technical Specifications (PDTS)

In Section 2.2 of the LAR Enclosure, Holtec stated that the proposed changes would revise certain requirements contained within the PDTS to reinstate requirements that are necessary for power operation and revise or remove requirements that would no longer be applicable. Thus, the NRC staff's approach for its review of the proposed changes to the PNP PDTS focused on evaluating whether the Holtec's proposed changes would appropriately reinstate the PNP TS, with some exceptions, to those operational TS that were in effect just prior to the docketing of the 10 CFR 50.82(a)(1) certifications, such that they are consistent with the PNP power operations TS previously approved by the NRC staff. To facilitate this review approach, the NRC staff compared the PNP TS sections proposed to be reinstated (with some exceptions) for consistency with the previous NRC-approved PNP TS that were in effect just prior to the issuance of PNP License Amendment No. 272. License Amendment No. 272 became effective following the docketing of the 10 CFR 50.82(a)(1) certifications.

The proposed changes to the PNP PDTS are described in LAR Enclosure Section 3.2.2, "Proposed Changes to the Permanently Defueled Technical Specifications," and shown in attachments to the Enclosure: Attachment 1 (mark-up pages) and Attachment 2 (retyped pages). The LAR Enclosure Section 3.2.2 is arranged in a table format that identifies the TS section, summarizes the proposed changes, and provides a basis for the proposed changes.

Holtec also submitted supplemental information to provide further clarity and administrative corrections to the initial LAR TS documentation. For example, the supplemental information (not all inclusive):

- Renumbered the LAR Enclosure Table of Contents and associated Section from 3.1.2 to 3.2.2 to correct an editorial error.

- Inserted a paragraph into LAR Enclosure Section 3.2.1 to document that there were no non-conservative TS associated with the PNP TS prior to or since docketing the 10 CFR 50.82(a)(1) certifications.

- Updated the LAR Enclosure Section 3.2.2 basis text for TS Sections 1.2, 1.3, 1.4, 2.0, 3.0, 5.6.2, and 5.6.3, to reference Amendment No. 272, which provides information that supports and clarifies the basis for the proposed changes to the PDTS.

- Updated the LAR Enclosure Section 3.2.2 basis text for TS 3.7.14, TS 3.7.15, and TS 3.7.16 to state in part, that these TSs are reinstated to what was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications.

- Updated LAR Enclosure Attachments 1 and 2 to correct editorial errors (e.g., spelling) and improve formatting (e.g., table headings) consistent with NUREG-1432, Revision 5 guidance.

The NRC staff's review of Holtec's proposed changes to the PNP PDTS is provided in the following Section 3.4 subsections of this SE.

3.4.1 Appendix A Title Page:

Renewed Facility Operating License DPR-20, Appendix A contains the PNP Technical Specifications.

The title page for Appendix A currently states, in part:

APPENDIX A
PERMANENTLY DEFUELED
TECHNICAL SPECIFICATIONS

Holtec proposed to change the Appendix A title page to state, in part:

APPENDIX A
TECHNICAL SPECIFICATIONS

The NRC staff reviewed Holtec's proposed TS changes described in LAR Enclosure Section 3.2.2 for "Appendix A Title Page," and as shown in attachments to the enclosure. The NRC staff finds that the proposed change is acceptable because it is consistent with TS title pages for plants that are operating and not permanently defueled, and it does not alter TS requirements.

3.4.2 Appendix A Table of Contents

Holtec proposed to remove Appendix A Table of Contents (TOC) from the PNP TS and place it under licensee control. The TOC provides information as to where (i.e., page number) specific TS sections can be found and does not contain any technical information.

The NRC staff reviewed the proposed TS changes described in LAR Enclosure Section 3.2.2 for Appendix A TOC. The NRC staff finds that removal of the TOC from the PNP TS is acceptable because inclusion of a TOC is not required by 10 CFR 50.36 and removing the TOC does not alter TS requirements.

3.4.3 TS Section 1.0, "Use and Application"

TS Section 1.0, "Use and Application" is comprised of the following specifications:

- TS Section 1.1, "Definitions," provides defined terms that are applicable throughout the TS and TS Bases.

- TS Section 1.2, "Logical Connectors," explains how the arrangement of logical connectors constitutes logical conventions with specific meanings.

- TS Section 1.3, "Completion Times," establishes the Completion Time convention and provides guidance for its use.

- TS Section 1.4, "Frequency," defines the proper use and application of Frequency requirements.

Holtec described the proposed changes in LAR Enclosure Section 3.2.2 for TS Sections 1.1, 1.2, 1.3, and 1.4. In this LAR enclosure section, Holtec explained that the TSs within TS Section 1.0 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The NRC staff notes Holtec has submitted a separate license amendment request in the Administrative Controls TS Amendment (ML24040A089) that proposed PNP PDTS changes to address, in part, staffing requirements no longer necessary to support a power operations plant (e.g., TS 1.1 definitions for CERTIFIED FUEL HANDLER and NON-CERTIFIED OPERATOR are no longer applicable for a plant authorized to operate and will be deleted). As previously discussed, the staff has separately reviewed and approved the Administrative Controls TS Amendment (ML25157A107). Thus, although a separate LAR proposes changes to information in the same PNP PDTS section, the NRC staff is concurrently issuing its approval of both LARs, and including conditions that would ensure that they are implemented at the same time.

The NRC staff reviewed Holtec's proposed TS changes described in LAR Enclosure Section 3.2.2 for TS Sections 1.1, 1.2, 1.3, and 1.4, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272, and are necessary to support the resumption of power operations at PNP. Additionally, the NRC staff finds that these “Use and Application” TS sections are proposed to be reinstated because they are necessary to support other TSs proposed to be reinstated as part of this LAR that are evaluated in the remaining subsections of SE Section 3.4. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 1.0 meet the requirements of 10 CFR 50.36(c), and are therefore acceptable.

3.4.4 TS Section 2.0, “Safety Limits”

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, “TS Section 2.0, Safety Limits (SLs).” In this LAR enclosure section, Holtec explained that the TSs within TS Section 2.0 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 2.0 contains safety limits that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity from the reactor core and the primary coolant system pursuant to 10 CFR 50.36(c)(1).

The NRC staff reviewed Holtec’s proposed TS changes described in this LAR section and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain safety limits that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 2.0 meet the requirements of 10 CFR 50.36(c)(1), and are therefore acceptable.

3.4.5 TS Section 3.0, “Limiting Conditions for Operation (LCO) Applicability”

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, “TS Section 3.0, Limiting Conditions for Operation (LCO) Applicability.” In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.0 (LCO Applicability) are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed revisions to TS Section 3.0 (LCO Applicability) contain the general requirements applicable to all LCOs and apply at all times unless otherwise stated in a TS.

The NRC staff reviewed Holtec’s proposed TS changes described in this LAR section and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain limiting conditions for operations that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.0 (LCO Applicability) meet the requirements of 10 CFR 50.36(c)(2), and are therefore acceptable.

3.4.6 TS Section 3.0, “Surveillance Requirement (SR) Applicability”

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, “TS Section 3.0, Surveillance Requirement (SR) Applicability.” In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.0 (SR Applicability) are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed revisions to TS Section 3.0 (SR Applicability) contain the general requirements applicable to all SRs and apply at all times unless otherwise stated in a TS.

The NRC staff reviewed Holtec's proposed TS changes described in this LAR section and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain surveillance requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.0 (SR Applicability) meet the requirements of 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.7 TS Section 3.1, "Reactivity Control Systems"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.1 Reactivity Control Systems." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.1 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications, with one administrative revision. The revision pertains to TS 3.1.4, "Control Rod Alignment." Specifically, the note in SR 3.1.4.3 excepted a control rod drive from the SR for a specific period of time (i.e., fuel cycle 25, which began on October 19, 2015). The cycle referenced in the note has passed and all control rod drives are subject to the SR going forward. Therefore, Holtec explained that the note was no longer needed and was not reinstated. The proposed TS Section 3.1 contains requirements to assure and verify operability of reactivity control systems. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that, apart from the revision discussed above, the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. The NRC staff also finds that removing the note is acceptable because the note has expired, and its removal does not alter TS requirements. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.1 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.8 TS Section 3.2, "Power Distribution Limits"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.2, Power Distribution Limits." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.2 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.2 contains power distribution limits that provide assurance that fuel design criteria are not exceeded. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.2 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.9 TS Section 3.3, "Instrumentation"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.3, Instrumentation." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.3 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.3 contains operability requirements for sensing and control instrumentation required for safe operation of the facility. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.3 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.10 TS Section 3.4, "Primary Coolant System (PCS)"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.4, Primary Coolant System (PCS)." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.4 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.4 contains requirements that provide for appropriate control of process variables, design requirements, or operating restrictions needed for appropriate functional capability of PCS equipment required for safe operation of the facility. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.4 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.11 TS Section 3.5, "Emergency Core Cooling Systems (ECCS)"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.5, Emergency Core Cooling Systems (ECCS)." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.5 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.5 contains requirements that provide for appropriate functional capability of ECCS equipment required for mitigation of DBAs or transients to protect the integrity of a fission product barrier. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.5 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.12 TS Section 3.6, "Containment Systems"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.6, Containment Systems." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.6 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.6 contains requirements that assure the integrity of the containment, depressurization and cooling systems, and containment isolation valves. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.6 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.13 TS Section 3.7, "Plant Systems"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.7, Plant Systems." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.7 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.7 contains requirements for the appropriate functional capability of plant equipment required for safe operation of the facility, including requirements that apply when the facility is in a defueled condition. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.7 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.14 TS Section 3.8, "Electrical Power Systems"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.8, Electrical Power Systems." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.8 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.8 contains requirements that provide for appropriate functional capability of plant electrical equipment required for safe operation of the facility. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.8 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.15 TS Section 3.9, "Refueling Operations"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 3.9, Refueling Operations." In this LAR enclosure section, Holtec explained that the TSs within TS Section 3.9 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed TS Section 3.9 contains requirements that provide for appropriate functional capability of plant electrical equipment required for safe operation of the facility. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 3.9 meet the requirements of 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3), and are therefore acceptable.

3.4.16 TS Section 4.0, "Design Features"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 4.0, Design Features." In this LAR enclosure section, Holtec explained that TSs within TS Section 4.0 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. The proposed changes to TS Section 4.0 contain requirements for reactor fuel assemblies and control rods in the reactor core, which are necessary in an operating plant. In addition, the proposed changes contain requirements for new fuel storage to address new fuel assemblies needed for reactor reload activities.

The NRC staff reviewed Holtec's proposed changes described in this LAR section and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 4.0 meet the requirements of 10 CFR 50.36(c)(4), and are therefore acceptable.

3.4.17 TS Section 5.0, "Administrative Controls"

Holtec described the proposed changes to this TS section in LAR Enclosure Section 3.2.2, "TS Section 5.0, Administrative Controls." In this LAR enclosure section, Holtec explained that TSs within TS Section 5.0 are proposed for reinstatement in their entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications. Specifically, in this LAR, the proposed changes reinstate applicable TSs within TS Section 5.5, "Programs and Manuals," and TS Section 5.6, "Reporting Requirements." The NRC staff notes Holtec has submitted a separate license amendment request in the Administrative Controls TS Amendment that proposed PNP PDTS changes to address, in part, the remaining TS administrative control requirements within TS Section 5.1, "Responsibility;" TS 5.2, "Organization;" TS 5.3, "Plant Staff Qualifications;" and TS 5.4, "Procedures." As previously discussed, the staff has separately reviewed and approved the Administrative Controls TS Amendment. Thus, although a separate LAR proposes changes to information in the same PNP PDTS section, the NRC staff is concurrently issuing its approval of both LARs, and including conditions that would ensure that they are implemented at the same time. Holtec also submitted supplemental information "… to provide further clarity and administrative corrections …" to this TS Section.

The NRC staff reviewed Holtec's proposed changes described in this LAR section, as supplemented, and as shown in the attachments to the enclosure. Based on its review, the NRC staff finds that the proposed TS changes contain requirements that are necessary to support the resumption of power operations at PNP. In addition, the staff finds that the proposed changes are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the issuance of Amendment No. 272. Furthermore, the NRC staff finds the proposed changes to this PNP TS section submitted as supplemental information are consistent with NUREG-1432 (STS) guidance and do not substantively alter TS requirements. Based on these findings, the NRC staff concludes that the proposed changes to PNP TS Section 5.0 meet the requirements of 10 CFR 50.36(c)(5), and are therefore acceptable.

3.4.18 TS Bases

Holtec described the proposed changes to the PNP TS Bases in LAR Enclosure Section 3.2.4, “Proposed Changes to the PNP Technical Specification Bases.” In this LAR enclosure section, Holtec explained that the TS Bases are provided for information only with markups shown in LAR Enclosure Attachment 3. In addition, Holtec explained that changes to the TS Bases will be incorporated in accordance with PNP TS 5.5.12, “Technical Specifications (TS) Bases Control Program.” Holtec also submitted supplemental information that identified changes to LAR Enclosure Attachment 3 to improve reader clarity, correct editorial errors, and include information to conform the TS Bases to that which existed prior to the docketing of the 10 CFR 50.82(a)(1) certifications.

The regulation at 10 CFR 50.36(a)(1) states that a summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TSs. Consistent with 10 CFR 50.36(a)(1), Holtec submitted corresponding TS Bases changes that provide the reasons for the proposed TSs changes. The NRC staff concludes that the TS Bases changes provided describe the bases for the affected TSs and follow the “Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors” (58 FR 39132).

3.4.19 Overall TS Conclusion

Based on the considerations discussed in Section 3.4 of this SE, the NRC staff concludes that the proposed changes to reinstate the PNP power operation TSs are acceptable because they are consistent with the previous NRC-approved PNP operational TS that were in effect at PNP just prior to the docketing of the 10 CFR 50.82(a)(1) certifications which triggered implementation of Amendment No. 272 and, other than the editorial and administrative changes noted above, are necessary for the resumption of power operations at PNP. Accordingly, the staff finds that the proposed changes meet the requirements in 10 CFR 50.36.

3.5 Proposed Changes to the Environmental Protection Plan

In Section 3.2.3 of the LAR Enclosure, Holtec stated that all changes to the EPP proposed in this LAR are made solely to more accurately reflect the PNP plant after it resumes power operations and to ensure the terminology used in the EPP is consistent with that used in the plant license. The NRC staff’s review approach for the proposed changes to the PNP EPP focused on evaluating whether Holtec’s proposed changes would appropriately reinstate the PNP EPP to the version of the PNP EPP that was in effect just prior to the docketing of the 10 CFR 50.82(a)(1) certifications, such that it is consistent with the previous NRC-approved PNP power operations EPP. To facilitate this review approach, the NRC staff compared the PNP EPP proposed to be reinstated for consistency with the previous NRC-approved PNP EPP that was revised by issuance of PNP License Amendment No. 272.

Holtec described the proposed changes to the PNP EPP in LAR Enclosure Section 3.2.3, “Proposed Changes to RFOL Appendix B, Environmental Protection Plan,” and shown in attachments to the Enclosure: Attachment 1 (mark-up pages) and Attachment 2 (retyped pages). In this LAR enclosure section, Holtec explained that all proposed changes to the EPP are made to reflect PNP after it resumes power operation. Holtec stated, “[t]he proposed changes do not alter the obligations in the environmental area, including, as appropriate, requirements for reporting and keeping records of environmental data, and any conditions and monitoring requirement for the protection of the nonaquatic environment. As such, the changes to the EPP proposed by this LAR are administrative changes only." In addition, Holtec explained that the EPP is proposed for reinstatement in its entirety to that which was in effect prior to the docketing of the 10 CFR 50.82(a)(1) certifications.

The specific changes proposed in the LAR involved several instances of changing the word "facility" to the word "plant" or "station;" changing language about the storage of spent fuel and maintenance of the plant to discuss construction and operation of the facility; returning references to power level and plant operation; and removing a reference to previous plant operation. These changes are administrative in nature to reflect that PNP will be in an operating condition and to more closely align the terminology in the EPP with that in the power operations technical specifications.

The NRC staff reviewed Holtec's proposed changes described in this LAR section and as shown in the LAR attachments to the enclosure. The NRC staff finds that the proposed changes are consistent with the previous NRC-approved PNP power operation EPP that was in effect just prior to the issuance of Amendment No. 272 and are necessary to support the resumption of power operations at PNP. Based on these findings, the NRC staff finds that the proposed changes meet the requirements of 10 CFR 50.36b, and are therefore acceptable.

3.6 Proposed Changes to the Physical Security Plan

PNP License Condition 2.E states, in part, that the licensee "shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revision to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Palisades Nuclear Plant Physical Security Plan."

Regulations in 10 CFR 73.55(c) and (d) state the requirements for licensees to establish, maintain, and implement a PSP to meet the requirements of 10 CFR 73.55, as well as Appendix B, "General Criteria for Security Personnel," and Appendix C, "Licensee Safeguards Contingency Plans," of 10 CFR Part 73. The licensee must show establishment and maintenance of a security organization, the use of security equipment and technology, the training and qualification of security personnel, the implementation of predetermined response plans and strategies, and the protection of digital computer and communication systems and networks. The licensee must also have a management system for development, implementation, revision, and oversight of security implementing procedures. The licensee will document the approval process for implementing security procedures. Security plans must describe how the licensee will implement Commission requirements and those site-specific conditions that affect implementation as required by 10 CFR 73.55(c)(1)(i).

Before and after the certifications of permanent cessation of reactor operations and permanent fuel removal from the reactor, PNP maintained the PNPPSP per the appropriate licensing conditions and NRC requirements. The current PNPPSP in place at PNP is Revision 21, dated September 3, 2024 (ML24330A136; non-public, SGI). PNP's Security Plan information is withheld from public disclosure in accordance with the provisions of 10 CFR 73.21.

For clarification, by letter dated October 28, 2004 (ML043170657), the last Commission-approved version of the PNPPSP was Revision 0, dated October 18, 2004 (not publicly available, SGI). Since the NRC staff's approval of PNPPSP, Revision 0, and while PNP was an operating reactor, PNP made changes to the PNPPSP from Revision 0 to Revision 16 in accordance with 10 CFR 50.54(p)(2). The NRC staff determined that changes in Revisions 1 through Revision 15 of the PSP conformed with 10 CFR 50.54(p).

PNP submitted PNPPSP, Revision 16 to the NRC on June 11, 2014. The NRC staff completed the review of Revision 16 on August 25, 2014 (ML14248A204; non-public). The NRC staff determined that changes in Revision 16 of the PNPPSP conformed with 10 CFR 50.54(p).

Since the certifications of permanent cessation of reactor operations and permanent fuel removal from the reactor, PNP made five additional changes to the PNPPSP pursuant to 10 CFR 50.54(p)(2) to reflect the reduced risks of a reactor in decommissioning (ML22231B022, ML22258A240, ML22353A493, ML23094A131, ML24330A136). The current physical security plan in place at PNP is Revision 21, dated September 3, 2024 (ML24330A136; non-public, SGI). The NRC staff reviewed the changes in each of the licensee's five PNPPSP decommissioning submittals and determined that the changes conformed with 10 CFR 50.54(p). The NRC staff recommended no further regulatory actions.

As stated in the LAR, PNP plans to update the current PNPPSP, Revision 21, following the requirements of 10 CFR 50.54(p)(2), to the docketed version that was in effect before the 10 CFR 50.82(a)(1) certifications were submitted, which was PNPPSP, Revision 16 (not publicly available, SGI). However, the LAR also proposes to amend the license to authorize operation of a power reactor. Licensed operation of a power reactor requires acceptable power reactor security plans under 10 CFR 50.34(c), 10 CFR 50.34(d) and 10 CFR 73.55. Therefore, to make the required findings for the proposed reauthorization of power operations at PNP as part of this LAR, the NRC staff reviewed the licensee's proposal to update the Security Plan to PNPPSP, Revision 22, which incorporates retained changes made during decommissioning.

The NRC staff's review of PNPPSP, Revision 22, focused on determining whether the format and content conformed with the NRC-endorsed NEI 03-12 (ML11301A066). SRP Chapter 13, Section 13.6.1, states, "A security plan conforming to NEI 03-12 is considered acceptable if it contains sufficiently detailed information addressing how all regulatory requirements will be met, captures the licensing bases for a physical protection program which meets the performance and prescriptive requirements set forth in 10 CFR Part 73, and supports the overall reasonable assurance safety finding that the issuance of a license and subsequent operations will not be an unacceptable risk to public health and safety … or inimical to common defense and security."

3.6.1 Review of the Palisades Nuclear Plant Physical Security Plan

The NRC staff reviewed Holtec's proposed changes described in the LAR and as shown in the LAR attachments to the enclosure. Additionally, the NRC staff reviewed the proposed changes to PNPPSP, Revision 16, as described above and documented in the submitted PNPPSP, Revision 22. Based on its review, the NRC staff finds that PNPPSP, Revision 22, with the changes listed above, is consistent with PNPPSP, Revision 16, which was in effect at PNP just prior to docketing the 10 CFR 50.82(a)(1) certifications. In addition, the NRC staff reviewed the content of PNPPSP, Revision 22 and compared it to the format and content in NEI 03-12. The staff finds that the PNPPSP was consistent with the format and content of NEI 03-12 and describes how PNP would meet the requirements set forth in 10 CFR 73.55(a) through (r). Therefore, the NRC staff concludes that PNPPSP, Revision 22 is acceptable.

3.6.2 Review of the Training and Qualification Plan

The NRC staff reviewed Holtec's proposed changes described in the LAR and as shown in the LAR attachments to the enclosure. Additionally, the NRC staff reviewed the proposed changes to the Training and Qualification Plan (T&QP), Revision 16, as described above and documented in the submitted T&QP, Revision 22. Based on its review, the NRC staff finds that T&QP, Revision 22, with the changes listed above, is consistent with T&QP, Revision 16, which was in effect at PNP just prior to docketing the 10 CFR 50.82(a)(1) certifications. In addition, the NRC staff reviewed the content of the T&QP, Revision 22 and compared it to the format and content in NEI 03-12. The staff finds that the T&QP was consistent with the format and content of NEI 03-12 and describes how PNP would meet the requirements set forth in 10 CFR Part 73, Appendix B. Therefore, the NRC staff concludes that T&QP, Revision 22 is acceptable.

3.6.3 Review of the Safeguards Contingency Plan

The NRC staff reviewed Holtec's proposed changes described in the LAR and as shown in the LAR attachments to the enclosure. Additionally, the NRC staff reviewed the proposed changes to SCP, Revision 16, as described above and documented in the submitted SCP, Revision 22. Based on its review, the NRC staff finds that SCP, Revision 22, with the changes listed above, is consistent with SCP, Revision 16, which was in effect at PNP just prior to docketing the 10 CFR 50.82(a)(1) certifications. In addition, the NRC staff reviewed the content of SCP, Revision 22 and compared it to the format and content in NEI 03-12. The staff finds that the SCP was consistent with the format and content of NEI 03-12 and describes how PNP would meet the requirements set forth in 10 CFR Part 73, Appendix C. Therefore, the NRC staff concludes that SCP, Revision 22 is acceptable.

3.6.4 Review of the ISFSI Security Plan

The NRC staff reviewed Holtec's proposed changes described in the LAR and as shown in the LAR attachments to the enclosure. Additionally, the NRC staff reviewed the proposed changes to ISFSI Security Plan, Revision 16, as described above and documented in the submitted ISFSI Security Plan, Revision 22. Based on its review, the NRC staff finds that ISFSI Security Plan, Revision 22, with the changes listed above, is consistent with ISFSI Security Plan, Revision 16, which was in effect at PNP just prior to docketing the 10 CFR 50.82(a)(1) certifications. In addition, the NRC staff finds that the ISFSI Security Plan, Revision 22 describes how PNP would meet the requirements set forth in 10 CFR 73.55 (a) through (r), with the conditions and exceptions specified in 10 CFR 72.212(b)(9). Therefore, the NRC staff concludes that ISFSI Security Plan, Revision 22 is acceptable.

3.6.5 Overall PNP Physical Security Plan Conclusion

The NRC staff's review of the PNPPSP, Revision 22, focused on ensuring the plans' format and content conformed with NEI 03-12 and contained the necessary programmatic elements to provide reasonable assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unacceptable risk to the public health and safety.

The NRC staff concluded that the PNPPSP, Revision 22, conforms with the format and content of the NRC-endorsed NEI 03-12. The plan, if effectively implemented, will provide the required reasonable assurance that the PNP security force can protect the plant against the design-basis threat of radiological sabotage. The burden to effectively implement these plans remains with the licensee. Effective implementation depends on the facility procedures and practices, that were not part of this review, which the licensee develops to satisfy the programmatic elements of the PNP PSP, T&QP, SCP, and ISFSI Security Plan.

The PNP target set analysis and the site's protective strategy are in facility implementing procedures, which were not subject to NRC staff review as part of this LAR, and are subject to future NRC review and inspection in accordance with 10 CFR 73.55(c)(7)(iv) and paragraph II.B.5(iii) in Appendix C of 10 CFR Part 73.

3.7 Technical Conclusion

Based on the considerations discussed in Section 3 of this SE, the NRC staff concludes that the proposed changes to reinstate the PNP power operation RFOL, TS, EPP, and PSP are acceptable because they are consistent with the previous NRC-approved PNP operational RFOL. TS. EPP, and PSP that were in effect at PNP just prior to the docketing of the 10 CFR 50.82(a)(1) certifications which triggered implementation of Amendment No. 272 and, other than the editorial and administrative changes noted in the SE, are necessary to support the resumption of power operations at PNP.

As described in Section 2.3 of this SE, concurrent with the issuance of this amendment, the NRC staff is issuing its approval of the Exemption Request (ML25163A182) to allow for a one-time rescission of the PNP docketed 10 CFR 50.82(a)(1) certifications to remove the restriction that prohibits operation of the PNP reactor and emplacement and retention of fuel into the PNP reactor vessel. As such, this license amendment is effective upon the licensee's submittal of a request to rescind the 10 CFR 50.82(a)(1) certifications.

Accordingly, the NRC staff finds that the proposed changes meet the applicable NRC regulatory requirements and provide reasonable assurance that the changes will be protective of public health and safety, the environment, and the common defense and security. Therefore, the approval of this LAR will continue to ensure that the PNP licensing basis is adequate to safely support the plant as it transitions to operations.

4.0 DISPOSITION OF PUBLIC COMMENTS

On August 7, 2024 (89 FR 64486), the NRC staff published a "Notice of Consideration of Issuance of Amendments to Facility Operating License, Opportunity to Comment, Request a Hearing, and Petition for Leave to Intervene" in the *Federal Register* associated with the proposed amendment requests. In accordance with the requirements in 10 CFR 50.91, "Notice for public comment; State consultation," the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determinations. Public comments were received regarding the proposed amendment (ML23348A148).

The NRC staff reviewed these comments and determined that the specific comments pertain to (1) appropriate methods for updating the PNP UFSAR to reflect the operating licensing basis; (2) justification of the current plant licensing basis against the NRC's General Design Criterion; (3) the need to submit an analysis of steam generator tube integrity; (4) further explanation of the regulatory framework related to the resumption of power operations at a decommissioning plant; and (5) consideration of potential process issues in the application of 10 CFR 50.12, "Specific exemptions." Because the issues discussed in the public comments do not specifically pertain to the proposed NSHC determination and thus go beyond the scope of the comment opportunity provided in the *Federal Register* notice, these comments are not further addressed in this SE. However, the staff notes that the commenter has raised similar issues through other NRC processes related to the potential restart of PNP, including a request for hearing, several requests made pursuant to 10 CFR 2.206, "Requests for action under this subpart," and a request for rulemaking under 10 CFR 2.802, "Petition for rulemaking." For example, the commenter raised similar issues in a hearing request filed on this LAR (ML24253A185). The Atomic Safety and Licensing Board denied the hearing request on March 31, 2025 (ML25090A164). Some of the other processes are still underway and the outcome of each will be documented in accordance with the associated process.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC staff's proposed no significant hazards consideration determination was published in the *Federal Register* on August 7, 2024 (89 FR 64486). On October 7, 2024, the NRC received two initial hearing requests on this LAR from: 1) Mr. Alan Blind on behalf of himself and Bruce Davis, Karen Davis, Jody Flynn, Thomas Flynn, Christian Moevs, Dianne Ebert, Mary Huffman, and Chuck Huffman, filed on September 9, 2024; and 2) Beyond Nuclear, Don't Waste Michigan, Michigan Safe Energy Future, Three Mile Island Alert, and Nuclear Energy Information Service (collectively, Petitioning Organizations). On March 3, 2025, the Petitioning Organizations filed a motion for leave to file new and amended contentions based on the publication of the Draft Environmental Assessment and Finding of No Significant Impact. On March 31, 2025 (ML25090A164), the Atomic Safety and Licensing Board (the Board) issued a Memorandum and Order denying both initial hearing requests. On April 25, 2025 (ML25115A265), the Petitioning Organizations appealed the Board's decision on their initial hearing request. On June 20, 2025 (ML25171A153), the Board issued an order denying the Petitioners Organization's motion for leave to file new and amended contentions. On July 15, 2025 (ML25196A132), the Petitioning Organizations appealed the Board's decision on the new and amended contentions. Both appeals are pending before the Commission.

Under the Atomic Energy Act of 1954, as amended, and the NRC's regulations, the NRC staff may issue and make an amendment immediately effective, notwithstanding the pendency before the Commission of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has made a final determination that no significant hazards consideration is involved.

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), Holtec provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

   Response: No.

   The proposed changes to the PNP RFOL, PDTS and EPP do not impact the design basis function of plant SSCs. The proposed changes do not affect accident initiators or precursors, nor do they alter design assumptions that could increase the probability or consequences of previously evaluated accidents.

Chapter 14 of the PNP UFSAR, Revision 35 describes the postulated design basis accidents and transient scenarios applicable to PNP during power operations. The UFSAR will be reinstated to reflect the docketed version (Revision 35) that was in effect prior to docketing the 10 CFR 50.82(a) certifications of permanent cessation of power operations and permanent removal of fuel at PNP. This will include restoration of the UFSAR, Revision 35 which includes previously evaluated accident analyses and safety classification of SSCs to support power operations at PNP. The proposed changes to the PDTS simply revise and/or add license conditions or specifications applicable to the PNP power operations licensing basis as previously evaluated in UFSAR, Revision 35. The proposed changes do not involve physical changes to the facility or in the procedures governing operation of the plant that were in effect prior to 10 CFR 50.82(a)(1) certifications.

The proposed addition / revision to TS definitions and rules of usage and application are those applicable to the reinstated PNP power operations technical specifications and have no impact on plant SSCs or the methods of operation of such SSCs.

The proposed reinstatement of PNP safety limits (SLs) and SL violations contain SLs that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity from the reactor core and the Primary Coolant System (PCS) pursuant to 10 CFR 50.36(c)(1). Since the proposed SLs are applicable to the power operations at PNP and provide protection of physical barriers to prevent uncontrolled radioactive release, they would not increase the probability or consequences of previously evaluated DBAs.

The reinstatement of TS Limiting Conditions for Operation and Surveillance Requirements that are related to the operation of the nuclear reactor or to the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated in the reinstated UFSAR. The safety functions involving core reactivity control, reactor heat removal, primary coolant system inventory control, and containment integrity are applicable at PNP as a power operation plant.

The proposed reinstatement of PNP design features contain features of the plant such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety pursuant to 10 CFR 50.36(c)(4). Since the proposed design features are applicable to the power operations at PNP and provide protection of important design features, they would not increase the probability or consequences of previously evaluated DBAs.

The addition and modification of provisions of the administrative controls of the PDTS and the non-radiological environmental protection requirements in the EPP do not affect any accidents applicable during power operation of the plant.

The probability of occurrence of previously evaluated accidents in the UFSAR is not increased since reinstatement of the previously approved licensing basis, including the RFOL, PDTS and EPP, is bounded by the reinstated analyses. Additionally, the proposed changes do not impact the function of plant structures, systems, or components.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes to the PNP RFOL, the PDTS, and the EPP have no impact on plant structures, systems or components. The proposed changes do not involve installation of new equipment or modification of existing equipment that could create the possibility of a new or different kind of accident. Hence, the proposed changes do not result in a change to the way the facility or equipment is operated in a manner which could cause a new or different kind of accident initiator to be created.

The addition of TS that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the plant will be operated within the previously approved POLB.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed amendment would modify the PNP RFOL and PDTS by adding the portions of the RFOL and TS that are credited in the accident analyses for the DBAs in the reinstated UFSAR. Postulated DBAs involving reactor operation are applicable because the plant will be in a power operation condition. These proposed changes impact operation of the facility and its response to transients or DBAs by reinstating requirements for equipment that is related to the operation of the nuclear reactor or to the prevention, diagnosis, or mitigation of reactor-related transients or accidents. The changes ensure that equipment required to respond to DBAs and transients described in the UFSAR remain capable of performing their safety function. No accident analyses or safety analyses acceptance criteria will be affected by the proposed changes.

Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

The NRC staff reviewed Holtec's no significant hazards consideration determination. Based on this review, the staff's evaluation of the underlying LAR as discussed above, and consideration of the public comments discussed in Section 4.0 of this safety evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment on June 3, 2025. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

In accordance with 10 CFR 51.30, 51.31, and 51.32, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment, as discussed in the NRC staff's environmental assessment and finding of no significant impact, issued on May 30, 2025 (90 FR 23071).

8.0 CONCLUSION

The Commission 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) there is reasonable assurance that 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.

Principal Contributors: Clint Ashley, NRR/STSB
Steven Sarver, NSIR

Date of Issuance: July 24, 2024

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT NO. 276 RE: AMENDMENT REQUEST TO REVISE RENEWED FACILITY OPERATING LICENSE AND PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS TO SUPPORT RESUMPTION OF POWER OPERATIONS (EPID L-2023-LLA-0174) DATED JULY 24, 2025

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DATE	06/06/2025	06/10/2025	06/11/2025	01/27/2025
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