Entergy

Entergy Nuclear Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213
Tel 601-368-5573

Ron Gaston
Director, Nuclear Licensing

10 CFR 50.90

NL-20-033

April 28, 2020

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Technical Specifications Proposed Change – Permanently Defueled Technical Specifications

Indian Point Nuclear Generating Station Unit No. 3
NRC Docket No. 50-286
Renewed Facility Operating License No. DPR-64

References:

1) Entergy Nuclear Operations, Inc. (Entergy) letter to U.S. Nuclear Regulatory Commission (NRC), "Notification of Permanent Cessation of Power Operations," (Letter No. NL-17-021) (ADAMS Accession No. ML17044A004), dated February 8, 2017

2) NRC letter to Entergy, "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Issuance of Amendment Nos. 292 and 267 RE: Changes to Technical Specification Sections 1.1, 'Definitions'; 4.0, 'Design Features'; and 5.0, 'Administrative Controls,' for a Permanently Defueled Condition (EPID L-2019-LLA-0081)," (ADAMS Accession No. ML20071Q717), dated April 10, 2020

3) Entergy Letter to NRC, "Proposed Technical Specifications (TS) Changes – Indian Point Nuclear Generating Unit 3 TS SR 3.7.7.2 and TS 3.7.6, Required Action A.1," (Letter No. NL-19-093) (ADAMS Accession No. ML19325E913), dated November 21, 2019

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.90, "Application for amendment of license or construction permit," Entergy Nuclear Operations, Inc. (Entergy) is proposing an amendment to Renewed Facility Operating License (FOL) DPR-64 for Indian Point Nuclear Generating Station Unit No. 3 (IP3). This proposed license amendment would revise the IP3 FOL and revise the Technical Specifications (TSs) in Appendix A to Permanently

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Defueled Technical Specifications (PDTS), the Environmental Technical Specification Requirements in Appendix B of the FOL, and the Inter-Unit Transfer Technical Specifications in Appendix C. The proposed changes are consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The proposed changes would revise certain requirements contained within the IP3 FOL and the Appendices A through C TSs and remove the requirements that would no longer be applicable after IP3 is permanently shut down and defueled.

In Reference 1, Entergy notified the U.S. Nuclear Regulatory Commission (NRC) that it has decided to permanently cease operations of IP3 by April 30, 2021. Once certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and they are docketed, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2).

The proposed changes to the IP3 FOL and TSs are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages in the FOL and the Appendix A TSs, where appropriate, to condense and reduce the number of pages without affecting the technical content. The Appendix A TSs Table of Contents is also accordingly revised. This license amendment request includes the changes to the IP3 TSs that were approved by the NRC in Reference 2, since these are required to support the permanently shut down and defueled condition. The changes proposed to the IP3 TSs in Reference 3 but not yet approved will not affect the IP3 TSs that will remain in the PDTS; thus, they are not included.

Entergy has reviewed the proposed amendment in accordance with 10 CFR 50.92 and concludes it does not involve a significant hazards consideration.

In accordance with 10 CFR 50.91, copies of this application, with the enclosure, are being provided to the New York State Department of Health and Emergency Management Agency.

The Enclosure to this letter provides a detailed description and evaluation of the proposed changes for IP3. Attachment 1 to the Enclosure contains a mark-up of the current FOL, TS, and TS Bases pages. The TS Bases pages are provided for information only. Attachment 2 to the Enclosure contains the retyped Renewed Facility License, PDTS, Appendices B and C TSs, and PDTS Bases pages in their entirety.

Entergy requests review and approval of this proposed license amendment by April 30, 2021, with a 90-day implementation period from the effective date of the amendment. The license amendment will not be implemented until the certifications required by 10 CFR 50.82(a)(1)(i) have been docketed in accordance with 10 CFR 50.82(a)(2), the decay time requirement established in the analysis of the Fuel Handling Accident in the Fuel Storage Building has been met, and the license amendment regarding the IP3 TSs affecting the administrative controls for the permanently defueled condition, which was approved by the NRC in Reference 2, has been implemented.

There are no new regulatory commitments made in this letter.

If you have any questions regarding this submittal, please contact Ms. Mahvash Mirzai, Manager, Regulatory Assurance, at 914-254-7714.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 28, 2020.

Respectfully,

Ron Gaston

RWG/cdm/std

Enclosure: Indian Point Nuclear Generating Station Unit No. 3 – Description and Evaluation of Proposed Changes

Attachments to Enclosure:

1. Indian Point Nuclear Generating Station Unit No. 3 – Mark-up of the Current Facility Operating License, Appendices A through C Technical Specifications, and Appendix A Technical Specifications Bases

2. Indian Point Nuclear Generating Station Unit No. 3 – Re-typed (Clean) Facility License, Appendix A Permanently Defueled Technical Specifications, Appendices B and C Technical Specifications, and Appendix A Permanently Defueled Technical Specifications Bases

cc: NRC Senior Project Manager, NRC NRR DORL
Regional Administrator, NRC Region I
NRC Senior Resident Inspector, Indian Point Energy Center
President and CEO, NYSERDA
New York State (NYS) Public Service Commission
NYS Department of Health – Radiation Control Program
NYS Emergency Management Agency

Enclosure

NL-20-033

Indian Point Nuclear Generating Station Unit No. 3
Description and Evaluation of Proposed Changes

Indian Point Nuclear Generating Station Unit 3
Description and Evaluation of Proposed Changes

1. SUMMARY DESCRIPTION

On February 8, 2017, Entergy Nuclear Operations, Inc. (Entergy) notified the U.S. Nuclear Regulatory Commission (NRC) that it would permanently cease power operations at Indian Point Nuclear Generating Station Unit 3 (IP3) no later than April 30, 2021 (Reference 1). Once certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and they are docketed, the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2).

This proposed license amendment would revise the IP3 Facility Operating License (FOL) and revise the Technical Specifications (TSs) in Appendix A of the FOL to Permanently Defueled Technical Specifications (PDTS), the Environmental Technical Specification Requirements in Appendix B of the FOL, and the Inter-Unit Fuel Transfer Technical Specifications in Appendix C. The proposed changes are consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The proposed changes would revise certain requirements contained within the IP3 FOL and TSs and remove the requirements that would no longer be applicable after IP3 is permanently shut down and defueled.

The license amendment will not be implemented until the certifications required by 10 CFR 50.82(a)(1)(i) have been docketed in accordance with 10 CFR 50.82(a)(2), the decay time requirement established in the analysis of the Fuel Handling Accident (FHA) in the Fuel Storage Building (FSB) has been met, and the license amendment regarding the IP3 TSs affecting the administrative controls for the permanently defueled condition, which was approved by the NRC in Reference 2, has been implemented.

2. DETAILED DESCRIPTION AND BASIS FOR THE CHANGES

This license amendment request modifies the IP3 FOL and revises the IP3 Appendix A TSs into PDTS, and IP3 Appendices B and C TSs to comport with a permanently shut down and defueled condition.

The proposed changes to the IP3 FOL and TSs are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages in the FOL and the Appendix A TSs, where appropriate, to condense and reduce the number of pages without affecting the technical content. The Appendix A TSs Table of Contents is also accordingly revised. This license amendment request assumes that the changes to the IP3 TSs that were approved by the NRC in Reference 2 have been implemented by the site. The changes proposed to the IP3 TSs in Reference 3 but not yet approved will not affect the IP3 TSs that will remain in the PDTS; thus, they are not included.

General Analysis Applicable to Proposed Change

The regulatory requirements related to the content of TSs are promulgated in 10 CFR 50.36, "Technical Specifications." As detailed in a subsequent section of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TSs. In a permanently defueled condition, the scope of equipment and parameters that must be included in the IP3 TSs is limited to those needed to address the remaining postulated design basis accidents (DBAs) that will remain applicable to IP3 in the permanently shut down and defueled condition, so that the consequences of the accident are maintained within acceptable limits.

Chapter 14 of the IP3 Updated Final Safety Analysis Report (UFSAR) describes the DBA and transient scenarios applicable to IP3 during power operations. During normal power operations, the forced inlet flow of water through the reactor coolant system (RCS) removes the heat from the reactor by generating steam. The RCS, operating at high temperatures and pressures, transfers this heat through the steam generator tubes to the secondary system. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the RCS. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems which could affect the reactor core.

After the certifications are submitted for permanent cessation of operations and removal of fuel from the IP3 reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and docketed pursuant to 10 CFR 50.82(a)(2), the majority of DBA scenarios postulated in the IP3 UFSAR will no longer be possible. The irradiated fuel will be stored in the Spent Fuel Pit (SFP) and the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped off site in accordance with the schedules provided in the Post Shut Down Decommissioning Activities Report (PSDAR) (Reference 4).

Chapter 14 of the IP3 UFSAR evaluates the safety aspects of the plant and demonstrates that the plant can be operated safely and that exposures from credible accidents do not exceed the guidelines of 10 CFR 100. This chapter is divided into three sections, each dealing with a different behavior category:

1. Core and Coolant Boundary Protection Analysis, Section 14.1 – The incidents presented in Section 14.1 generally have no offsite radiation consequences.

2. Standby Safeguards Analysis, Section 14.2 –The accidents presented in Section 14.2 are more severe and may cause the release of radioactive material to the environment.

3. Rupture of a Reactor Coolant Pipe, Section 14.3 – The accident presented in Section 14.3, the rupture of a reactor coolant pipe, is the worst-case accident and is the primary basis for the design of engineered safety features.

Safety analyses are evaluated against regulatory acceptance criteria and are integral of the plant's design and licensing basis. The safety analyses demonstrate the integrity of the fission product barriers, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents and transients. Systems, Structures, and Components (SSCs) that perform design basis functions are credited in the safety analyses for the purpose of mitigating the transient or accident.

A list of the IP3 UFSAR Chapter 14 transients and DBAs and a determination regarding whether the transient or accident applies to a permanently defueled condition is provided in Table 2-1.

Table 2-1 – IP3 DBAs and Events

UFSAR Section	Postulated Accident or Transient	Defueled Applicability
14.1.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition	Not Applicable
14.1.2	Uncontrolled Control Rod Assembly Withdrawal at Power	Not Applicable
14.1.3	Rod Assembly Misalignment	Not Applicable
14.1.4	Rod Cluster Control Assembly (RCCA) Drop	Not Applicable
14.1.5	Chemical and Volume Control System Malfunction	Not Applicable
14.1.6	Loss of Reactor Coolant Flow	Not Applicable
14.1.7	Startup of Inactive Reactor Coolant Loop	Not Applicable
14.1.8	Loss of External Electrical Load	Not Applicable
14.1.9	Loss of Normal Feedwater	Not Applicable
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions	Not Applicable
14.1.11	Excessive Load Increase Incident	Not Applicable
14.1.12	Loss of All AC Power to the Station Auxiliaries	Not Applicable
14.1.13	Startup Accidents without Reactor Coolant Pump Operation	Not Applicable
14.1.14	Startup Accident with a Full Pressurizer	Not Applicable
14.2.1	Fuel Handling Accidents	Applicable – Drop of a fuel assembly onto the floor of the spent fuel pit Not Applicable – Fuel Assembly Stuck Inside the Reactor Vessel Not Applicable – Fuel Assembly or Control Rod Cluster Dropped onto the Floor of the Reactor Cavity Not Applicable – Fuel Assembly Stuck in the Penetration Valve Not Applicable – Fuel Assembly Stuck in the Transfer Carriage or the Carriage Becomes Stuck Not Credible - Fuel

UFSAR Section	Postulated Accident or Transient	Defueled Applicability
		Cask Drop Accident Deemed to not be Credible – See discussion in (1) below
14.2.2	Accidental Release of Waste Liquid	Applicable – Dose dependent on volatilized components and is addressed in Section 14.2.3
14.2.3	Accidental Release - Waste Gas	Applicable
14.2.4	Steam Generator Tube Rupture	Not Applicable
14.2.5	Rupture of a Steam Pipe	Not Applicable
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCC Assembly Ejection)	Not Applicable
14.3	Loss-of-Coolant-Accidents	Not Applicable
Appendix 14B	Consequences of a Turbine Missile at Indian Point 3	Not Applicable

(1) Section 14.2.1 of the IP3 UFSAR states:

"As discussed in Section 9.12.4.3, Single Failure Proof Cranes [for] Spent Fuel [Casks], the fuel storage building crane's main hook that handles spent fuel casks has been upgraded to single-failure-proof in accordance with the applicable guidelines of NRC NUREG-0554 (Single-Failure-Proof Cranes for Nuclear Power Plants, May 1979) and the applicable requirements of American Society of Mechanical Engineers ASME NOG-1-2004, Rules [for] Construction of Overhead and Gantry Cranes (top Running Bridge, Multiple Girder) to support spent fuel cask handling activities, without the necessity of having to postulate the drop of a spent fuel cask. With the crane's main hook qualified as single-failure-proof, and when the [cranes] is used as part of a single-failure-proof handling system for critical lifts as discussed in NRC NUREG-0800, Revision 1 of Section 9.1.5, Overhead Heavy Load Handling Systems, [Sub-section III.4.C], a cask drop accident is not a credible event and need not be postulated..."

The analyzed accidents that remain applicable to IP3 in the permanently shut down and defueled condition are the FHA in the FSB, accidental release of waste liquid, and the accidental release of waste gas.

The additional discussion of the analyses of these events provided below is based on information from Calculation IP-CALC-19-00003, "Post-Permanent Shutdown Analyses of Fuel Handling, Waste Handling, and High Integrity Container Drop Accidents for Indian Point Units 2 and 3." This calculation includes:

- The results of an analysis of the FHA utilizing the Alternate Source Term (AST) methodology described in Regulatory Guide 1.183 that is provided in Calculation IP-CALC-11-00074, "AST Analysis of IP3 Fuel Handling Accident in the Fuel Storage Building without FSB Exhaust Fan Operation." This analysis concludes that the dose consequences of the FHA for the "Normal" case will remain within the licensing basis dose limits without crediting FSB ventilation, the station vent radiation monitors, Control Room isolation, and Control Room filtration assuming 84 hours of decay time following shut down.

- The determination of the dose consequences for a waste gas decay tank rupture accident using a 50,000 Curie (Ci) dose-equivalent Xe-133 waste gas tank activity limit without any credit for mitigating systems.

- An analysis of a High Integrity Container Drop event was performed. However, it is not credited as part of this PDTS license amendment request. It was performed for future purposes that are outside the changes requested by the PDTS license amendment request.

Analysis of the FHA in the FSB for the Permanently Shut Down and Defueled Condition

Concurrent with implementation of the PDTS, UFSAR Section 14.2.1 will be revised in accordance with 10 CFR 50.59 to reflect the results of the "Normal" case analyzed in Calculation IP-CALC-11-00074, as summarized in Calculation IP-CALC-19-00003. This FHA analysis utilizes the AST methodology and concludes that the dose consequences of the FHA in the FSB will remain within the licensing basis dose limits without crediting FSB ventilation, the station vent radiation monitors, Control Room isolation, and Control Room filtration assuming 84 hours of decay time following shut down. The FHA dose consequences in the IP3 Control Room, Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) were computed using the following:

- The methodology and assumptions in Regulatory Guide 1.183,

- Appropriate source terms, release pathways, and other assumptions, as described below,

- All fuel pins in the dropped fuel assembly are broken,

- Decay time experienced prior to fuel movement = 84 hours,

- SFP Water Depth = 23 feet,

- Post-accident atmospheric dispersion factors, and

- The NRC sponsored code RADTRAD, Rev. 3.03, was used to model the design basis FHA and estimate the dose consequences. The Control Room, EAB and LPZ doses in terms of Total Effective Dose Equivalent (TEDE) were calculated for the FHA.

Fission Product Inventory

The fission product inventory in the core is based on full power operation (3216 Megawatt-thermal (MWt) + 2% uncertainty, i.e., 3280.3 MWt). The core inventory of radionuclides of interest at 84 hours decay are shown in Table 2-2.

Table 2-2 – Core Inventories of Nuclides for Use in Radiological Design-Basis Applications (at 84 Hours of Decay)

Nuclide Halogens	Activity (Ci)	Nuclide Noble Gases	Activity (Ci)
I-130	3.41E+04	Kr-85m	5.62E+01
I-131	6.90E+07	Kr-85	1.11E+06
I-132	6.38E+07	Kr-87	0.00E+00
I-133	1.17E+07	Kr-88	0.00E+00
I-134	0.00E+00		
I-135	2.63E+04		
		Xe-131m	9.71E+05
		Xe-133m	2.78E+06
		Xe-133	1.36E+08
		Xe-135m	4.21E+03
		Xe-135	7.86E+05
		Xe-138	0.00E+00

Release Fractions and Composition

The fission product gap release fractions, for each radionuclide group for the FHA are shown below:

- I-131 0.12
- Kr-85 0.30
- Other iodines and noble gases 0.10

The iodine released from the assembly gap is assumed to be 99.85% elemental and 0.15% organic.

The overall SFP decontamination factor for iodines is 200.

A value of 285 for the SFP elemental iodine decontamination factor was calculated.

Control Room Dose Consequences

The Control Room modeling assumptions are:

- The Control Room χ/Q for 0-2 hours is 1.07E-03 seconds/m^3
- Since releases are assumed to be completed in the first 2 hours (Regulatory Guide 1.183), no additional time periods are presented
- Control Room volume: 47,200 ft^3
- Filtered makeup: 0 ft^3/min
- Filtered recirculation: 0 ft^3/min
- Unfiltered makeup: 1,500 ft^3/min
- Unfiltered inleakage: 700 ft^3/min
- The breathing rate was assumed to be 3.5E-04 m^3/second for the duration of the accident.

Offsite Dose Consequences

The offsite modeling assumptions are:

- Offsite atmospheric dispersion factors (χ/Q)
  - EAB χ/Q is 1.03E-03 seconds/m^3
  - LPZ χ/Q is 3.8E-04 seconds/m^3
- Breathing Rates
  - EAB for 0-2 hours is 3.5E-04 (consistent with Regulatory Guide 1.183).
  - LPZ for 0-8 hours is 3.5E-04 (consistent with Regulatory Guide 1.183).
  - LPZ for 8-24 hours is 1.75E-04
  - LPZ for 24-720 hours is 2.32E-04

Radiological Consequences

The radiological consequences of the postulated FHA are as follows:

Table 2-3 – AST FHA Results (at 84 Hours of Decay)

Location	TEDE Dose (rem)	Regulatory Limit (rem)
Control Room	4.91	5
EAB	5.7	6.3
LPZ	2.1	6.3

The calculated TEDE values to the Control Room, EAB, and LPZ are less than the limits set forth in 10 CFR 50.67 and Regulatory Guide 1.183.

In addition, after a decay time of at least 720 hours (30 days) prior to fuel movement, the analysis of the FHA results in an EAB TEDE dose of 0.47 rem, which is less than the Environmental Protection Agency (EPA) Protective Action Guideline recommended threshold for evacuation of 1 rem.

Accidental Release – Waste Gas

Section 14.2.3 of the IP3 UFSAR evaluates the accidental release of waste gas. Concurrent with implementation of the PDTS, this UFSAR section will be revised in accordance with 10 CFR 50.59 to reflect the results of Calculation IP-CALC-19-00003, "Post-Permanent Shutdown Analyses of Fuel Handling, Waste Handling, and High Integrity Container Drop Accidents for Indian Point Units 2 and 3." This calculation includes the determination of the dose consequences for a waste gas decay tank rupture accident using a 50,000 Ci dose-equivalent Xe-133 waste gas tank activity limit without any credit for mitigating systems.

The waste gas decay tanks receive the radioactive gases from the radioactive liquids from the various laboratories and drains processed by the waste disposal system. The 50,000 Ci dose-equivalent Xe-133 waste gas tank activity assumed in this calculation bounds the current Xe-133 dose-equivalent limit of 29,761 Ci, as well as the administrative Xe-133 dose-equivalent limit of 6,000 Ci.

Other tanks that contain waste gas during operations (the volume control tank and liquid holdup tank) were not considered in this analysis, since gaseous products from these liquid tanks are collected and compressed in the waste gas decay tanks for decay prior to release. Potential liquid waste releases are considered from these tanks; however, any liquid releases are retained in the building or sumps and only volatilized components would be released to the environment. These volatilized components are evaluated as part of the waste gas decay tank accident.

Calculation

The RADTRAD model for the waste gas decay tank accident was developed using the following inputs to model an instantaneous ground level release of 50,000 Ci of Xe-133.

- 3 RADTRAD compartments – Waste Gas Decay Tank (modeled as a volume of 1 ft^3), Environment, and Control Room (volume = 4.72E4 ft^3)
- Plant power level – 1 MWt – Nominal power level used to release activity of 50,000 Ci of dose equivalent Xe-133 from the associated RADTRAD nuclide inventory file
- Activity release fractions – noble gas fraction = 1.0 over a duration of 0.00001 hour – This represents an instantaneous release in the RADTRAD release file
- Assumed high flow rate from Waste Gas Decay Tank volume to the Environment of 1E4 ft^3/min to model an instantaneous release from the tank compartment to the environment
- Control Room Flow Rate
  - Unfiltered intake = 1,500 ft^3/min
  - Unfiltered inleakage = 700 ft^3/min
  - Outflow = 2,200 ft^3/min
  - Recirculation = 0 ft^3/min
- Offsite χ/Qs: EAB = 1.03E-03 seconds/m^3, LPZ = 3.8E-04 seconds/m^3

A bounding χ/Q was developed for a release from the waste gas decay tanks to the IP3 Control Room. The release point is assumed to be the centerline of the closest large waste gas decay tank. The following inputs were used to develop the Control Room χ/Q.

Table 2-4 – Control Room Atmospheric Dispersion Parameters (Waste Gas Analysis)

Parameter	Value
IP3 Control Room intake location (ft)	5783.75 North 1476.0 East
Center line location IP3 gas decay tank #31 (ft)	5841.25 North 1552.5 East
Lateral dispersion coefficient for a 30 meter release, stability class F	1.5
Vertical dispersion coefficient for a 30 meter release, stability class F	0.85

The wind speed was assumed to be 1 meter/second and the stability class was assumed to be 'F' for calculating a bounding atmospheric dispersion coefficient from the Primary Auxiliary Building (PAB) to the Control Room. These generic meteorology conditions were used to calculate a bounding atmospheric dispersion coefficient. An additional significant conservatism is the implicit assumption that the wind direction is directly toward the Control Room intake at all times.

The IP3 Control Room χ/Q also does not credit holdup of the activity in the PAB which would delay or disperse activity within the building.

Radiological Consequences

The calculated radiological consequences, following a waste gas decay tank rupture without credit for any mitigating systems or the PAB ventilation system post shutdown, are provided in Table 2-5.

Table 2-5 – Waste Gas Decay Tank Rupture Results with 50,000 Ci Dose Equivalent Xe-133 Limit per Tank

Location	Whole Body Dose (rem)	Limit (rem)
Control Room	0.77	5.0
EAB	0.30	0.5
LPZ	0.11	0.5

The radiological consequences following a waste gas decay tank rupture are less than the dose consequences following an FHA presented in Table 2-3. They are also less than the 10 CFR 50.67 limit of 5 rem TEDE to the Control Room operators and the 500 mrem EAB and LPZ dose limit following a waste gas tank accident.

Accidental Release of Waste Liquid

Section 14.2.2 of the IP3 UFSAR addresses the accidental release of waste liquid. It concludes: "The incipient hazard from these process or waste liquid releases is derived only from the volatilized components. The releases are described and their effects summarized in Section 14.2.3." Therefore, a separate liquid-specific release accident evaluation is not required to be performed with regard to removal of supporting systems such as PAB ventilation, station vent radiation monitors, Control Room isolation, and Control Room filtration.

3. REGULATORY EVALUATION

Indian Point Nuclear Generating Station Unit 3 proposes to modify the license conditions and the TSs from Appendices A through C as listed in the following tables. In addition, IP3 is providing a description and basis for each of the proposed changes.

Attachment 1 to this enclosure contains a mark-up of the current FOL, Appendices A, B, and C TSs and Appendix A TSs Bases pages. The proposed changes to the IP3 Appendix A TSs are considered a major rewrite. Thus, the IP3 Appendix A TSs and TSs Bases that are deleted in their entirety are identified as such, but the associated deleted pages are not included in Attachment 1 to this enclosure. In addition, the following administrative changes are not shown in the marked-up (Enclosure, Attachment 1) FOL, Appendix A TSs, and Appendix A TSs Bases pages, because they do not affect the technical content of the IP3 FOL or Appendix A TSs:

- Reformatting (margins, font, tabs, line spacing, etc.) content to create a continuous electronic file; and
- Renumbering of pages, where appropriate, to condense and reduce the number of pages.

Attachment 2 of this enclosure provides the re-typed IP3 Facility License, PDTS, PDTS Bases in their entirety, and the affected pages of the Appendices B and C TSs. Since the changes to the Appendix A TSs and TSs Bases are considered a major rewrite, revision bars are not used. It incorporates the changes to the IP3 Appendix A TSs approved by the NRC in Reference 2.

Proposed Changes to the IP3 Facility Operating License

License Title	
Current Title Renewed Facility Operating License	Proposed Title Renewed Facility ~~Operating~~ License
Basis	
The License Title is modified to eliminate the reference to "Operating." After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 1.B	
Current License Condition 1.B The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;	Proposed License Condition 1.B The facility will ~~operate~~***be maintained*** in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
Basis	
This license condition is revised to reflect a more accurate description of the future requirements. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, replacing the verb "operate" with the verb "be maintained" will provide accuracy regarding the possession-only 10 CFR Part 50 License.	

License Condition 2	
Current License Condition 2 Accordingly, Renewed Facility Operating License No. DPR-64, is hereby issued to ENIP3 and ENO to read as follows:	Proposed License Condition 2 Accordingly, Renewed Facility ~~Operating~~ License No. DPR-64, is hereby issued to ENIP3 and ENO to read as follows:
Basis	
This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 2.A	
Current License Condition 2.A This renewed license applies to the Indian	Proposed License Condition 2.A This renewed license applies to the Indian

Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by ENIP3 and operated by ENO. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "Final Facility Description and Safety Analysis Report," as supplemented and amended, and the Environmental Report, as amended.	Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by ENIP3 and ~~operated~~***maintained*** by ENO. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "~~Final Facility Description and~~***Defueled*** Safety Analysis Report~~,~~" as supplemented and amended, and the Environmental Report, as amended.
Basis	
This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). In addition, it is modified to reflect that a Defueled Safety Analysis Report will be prepared to address the permanently shut down and defueled condition.	

License Condition 2.B.(1)	
Current License Condition 2.B.(1)	Proposed License Condition 2.B.(1)
Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENIP3 to possess and use, and (b) ENO to possess, use, and operate, the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;	Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENIP3 to possess and use, and (b) ENO to possess~~,~~ ***and*** use~~, and operate~~ the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;
Basis	
This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 2.B.(2)	
Current License Condition 2.B.(2)	Proposed License Condition 2.B.(2)
ENO pursuant to the Act and 10 CFR Part 70, to receive, possess, and use, at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Facility Description and Safety Analysis Report, as supplemented and amended;	ENO pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~, at any time special nuclear material ***that was used*** as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation,~~ as described in the ~~Final Facility Description and~~***Defueled*** Safety Analysis Report, as supplemented and amended;

Basis
This license condition is revised to remove the authorization for receipt and use of special nuclear material (SNM) as reactor fuel, eliminate the reference to use of the SNM for reactor operations, and limit the possession of SNM to SNM "that was used" as reactor fuel at IP3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As such, IP3 has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel is necessary as IP3 currently possesses the reactor fuel that was used for the past operations of the reactor. In addition, it is modified to reflect that a Defueled Safety Analysis Report will be prepared to address the permanently shut down and defueled condition.

License Condition 2.B.(3)	
Current License Condition 2.B(3)	Proposed License Condition 2.B.(3)
ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time, any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;	ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time, any byproduct, source and special nuclear material as sealed neutron sources ***that were used*** for reactor startup, sealed sources ***that were used*** for ***calibration of*** reactor instrumentation and ***are used in the calibration of*** radiation monitoring equipment ~~calibration~~, and ***that were used*** as fission detectors in amounts as required;
Basis	
This license condition is revised to remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup and fission detectors. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that IP3 will no longer be authorized to operate. The deletion of the authorization to receive and use fission detectors is consistent with the fact that IP3 complies with the criteria of 10 CFR 50.68(b) in lieu of maintaining a monitoring system capable of detecting criticality in the spent fuel pit as described in Section 9.5 of the IP3 Updated Final Safety Analysis Report. The authorization to possess such sources previously used for reactor startup and fission detectors is retained. The continued authorization to possess neutron sources that were used for reactor startup and fission detectors is consistent with the safe storage of byproduct, source, and SNM. The use of sources for radiation monitoring will continue to be required. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). These changes are consistent with the permanently defueled condition.	

License Condition 2.B.(4)	
Current License Condition 2.B(4) ENO 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 without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus or components.	Proposed License Condition 2.B.(4) ENO 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 without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus or components~~.~~;
Basis	
This license condition is revised by making a grammatical correction. The ending period is replaced with a semi-colon.	

License Condition 2.B.(5)	
Current License Condition 2.B(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.	Proposed License Condition 2.B.(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~***that were*** produced by the operation of the facility.
Basis	
This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 2.C.(1), Maximum Power Level	
Current License Condition 2.C.(1) ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal (100% of rated power).	Proposed License Condition 2.C.(1) ***Deleted per Amendment [###]***
Basis	
This license condition is deleted in its entirety to reflect the permanently defueled condition of the facility. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 2.C.(2), Technical Specifications	
Current License Condition 2.C.(2) The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 266, are hereby incorporated in the renewed license. ENO shall operate the facility in accordance with the Technical Specifications.	Proposed License Condition 2.C.(2) The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. ~~266~~***###***, are hereby incorporated in the renewed license. ENO shall ~~operate~~ ***maintain*** the facility in accordance with the Technical Specifications.
Basis	
This license condition is revised to replace the verb "shall operate" with the verb "shall maintain" to better describe the permanently defueled condition. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 2.C.(3)	
Current License Condition 2.C.(3) (DELETED)	Proposed License Condition 2.C.(3) ***None***
Basis	
The historical reference to a deleted license condition is deleted in its entirety. This is an administrative change.	

License Condition 2.C.(4)	
Current License Condition 2.C.(4) (DELETED)	Proposed License Condition 2.C.(4) ***None***
Basis	
The historical reference to a deleted license condition is deleted in its entirety. This is an administrative change.	

License Condition 2.H	
Current License Condition 2.H ENO shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for Indian Point Nuclear Generating Unit No. 3 and as approved in NRC fire protection safety evaluations (SEs) dated September 21, 1973, March 6, 1979, May 2, 1980, November 18, 1982, December 30, 1982, February 2, 1984, April 16, 1984, January 7, 1987, September 9, 1988, October 21, 1991, April 20, 1994, January 5, 1995, and supplements thereto, subject to the following	Proposed License Condition 2.H ***Deleted per Amendment [###]***

provision: ENO may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.	
Basis	
This license condition is deleted to reflect the permanently defueled condition of the facility. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the fire protection program will be revised to take into account the decommissioning facility conditions and activities. Indian Point Nuclear Generating Station Unit 3 will continue to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment. This 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 no longer be applicable at IP3. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during facility decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shut down and defueled facility is not needed.	

License Condition 2.O	
<u>Current License Condition 2.O</u> Evaluation, status and schedule for completion of balance of plant modifications as outlined in letter dated February 12, 1983, shall be forwarded to the NRC by January 1, 1984.	<u>Proposed License Condition 2.O</u> ***Deleted per Amendment [###]***
Basis	
The license condition is deleted in its entirety. It refers to a historical obligation that was previously met. The removal of this license condition is an administrative change.	

License Condition 2.AA	
<u>Current License Condition 2.AA</u> The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications: 1. This amendment authorizes the relocation of certain Technical Specification	<u>Proposed License Condition 2.AA</u> ***Deleted per Amendment [###]***

requirements and detailed information to licensee-controlled documents...The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment. 2. The following is a schedule for implementing surveillance requirements (SRs)...	
Basis	
The license condition is deleted in its entirety. It refers to a historical license condition associated with the conversion to the Improved Standard Technical Specifications. This license condition was previously met. Thus, the removal of this license condition is an administrative change.	

License Condition 2.AB	
Current License Condition 2.AB With the reactor critical, Entergy shall maintain the reactor coolant system cold leg at a temperature (T_{cold}) greater than or equal to 525 °F. Entergy shall maintain a record of the cumulative time that the plant is operated with the reactor critical while T_{cold} is below 525 °F. Upon determination by Entergy that the cumulative time of plant operation with the reactor critical while T_{cold} is below 525 °F has exceeded one (1) year, Entergy must: (a) within one (1) month, inform the NRC, in writing, and (b) within six (6) months submit the results of an analysis of the impact of the operation with T_{cold} below 525 °F on the pressurized thermal shock reference temperature (RT_{pts}).	Proposed License Condition 2.AB ***Deleted per Amendment [###]***
Basis	
The license condition is deleted in its entirety. It refers to operations with the reactor critical. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 2.AD, CONTROL ROOM ENVELOPE HABITABILITY	
Current License Condition 2.AD Upon implementation of Amendment No. 239 adopting TSTF-448, Revision 3 (as supplemented), the determination of control room envelope (CRE) unfiltered air inleakage as required by Technical Specification (TS)	Proposed License Condition 2.AD ***Deleted per Amendment [###]***

Surveillance Requirement (SR) 3. 7.11.4, in accordance with TS 5.5.16.c.(i), the assessment of CRE habitability as required by TS 5.5.16.c.(ii), and the measurement of CRE pressure as required by TS 5.5.16.d, shall be considered met. Following implementation: (a) The first performance of SR 3.7.11.4, in accordance with TS 5.5.16.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 1, 2005, the date of the most recent successful tracer gas test, as stated in the June 28, 2005, letter response to Generic Letter 2003-01. (b) The first performance of the periodic assessment of CRE habitability, TS 5.5.16.c.(ii), shall be within the next 9 months since the time period since the most recent successful tracer gas test is greater than 3 years. (c) The first performance of the periodic measurement of CRE pressure, TS 5.5.16.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from June 18, 2007, the date of the most recent successful pressure measurement test.	
Basis	
This license condition is deleted in its entirety. The license condition defined requirements of TSTF-448 to assess the Control Room Envelope (CRE) Habitability at the specified frequencies for the first performance of the specific test, assessment, and measurement. This is a historical license condition, because the test, assessment, and measurement were completed in accordance with the schedule specified in the license condition.	

License Condition 2.AF.(2).C, CONTROL ROOM ENVELOPE HABITABILITY	
<u>Current License Condition 2.AF.(2).c</u> The licensee shall notify the NRC in writing within 30 days after having accomplished item (2)a above and include the status of those activities that have been or remain to be completed in item (2)b above.	<u>Proposed License Condition 2.AF.(2).c</u> ***None***
Basis	
This license condition is deleted in its entirety. It is an obligation to notify the NRC within a specified time period regarding License Renewal activities that were required to be completed prior to the period of extended operation. This is a historical license condition, because the license condition was met in accordance with the schedule specified in the license condition. The removal of this license condition is an administrative change.	

License Condition 3	
Current License Condition 3 This renewed license is effective as of the date of its issuance, and shall expire at midnight April 30, 2025.	Proposed License Condition 3 This renewed license is effective as of the date of its issuance, and ~~shall expire at midnight April 30, 2025.~~***until the Commission notifies the licensee in writing that the license is terminated***.
Basis	
After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, this license condition is revised to conform with 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession by Entergy until the Commission notifies the licensee in writing that the license is terminated.	

ATTACHMENTS AND DATE OF ISSUANCE	
Current Attachments and Date of Issuance Appendix A – Technical Specifications... Date of Issuance: September 17, 2018	Proposed Attachments and Date of Issuance Appendix A – ***Permanently Defueled*** Technical Specifications... Date of Issuance: ~~September 17, 2018~~***To Be Determined***
Basis	
The title of Appendix A is updated to reflect that the Technical Specifications will be retitled as the Permanently Defueled Technical Specifications. The date of issuance is modified to reflect the date that the NRC issues the PDTS which is yet to be determined. These are administrative changes.	

APPENDIX A TO FACILITY ~~OPERATING~~ LICENSE DPR-64	
Current Title FACILITY OPERATING LICENSE DPR-64... TECHNICAL SPECIFICATIONS AND BASES... Amendment No. 203	Proposed Title FACILITY ~~OPERATING~~ LICENSE DPR-64... ***PERMANENTLY DEFUELED*** TECHNICAL SPECIFICATIONS AND BASES... Amendment No. ~~203~~***###***
Basis	
The License Title is modified to rename the "Facility Operating License DPR-64" and the "Technical Specifications and Bases" as "Facility License DPR-64" and "Permanently Defueled Technical Specifications and Bases." These changes reflect the upcoming change in status regarding IP3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention	

of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). In addition, the amendment number is modified to reflect the amendment number associated with the issuance of the PDTS.

APPENDIX A, TECHNICAL SPECIFICATIONS, TABLE OF CONTENTS	
Current IP3 TS	Basis for Change
Table of Contents	The Table of Contents is modified to reflect the changes made below.

TECHNICAL SPECIFICATION SECTION 1.1, DEFINITIONS

Technical Specification 1.1, "Definitions," provides defined terms that are applicable throughout the TSs and TSs Bases. A number of the Definitions are proposed to be deleted, because they have no relevance to and no longer apply to the permanently defueled facility status.

Definition	Basis for Change
ACTUATION LOGIC TEST	This definition is proposed for deletion, because the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible in the permanently defueled condition.
AXIAL FLUX DIFFERENCE (AFD)	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term is no longer applicable since fuel will be permanently removed from the reactor core.
CHANNEL CALIBRATION	This definition is proposed for deletion, because the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible in the permanently defueled condition.
CHANNEL CHECK	This definition is proposed for deletion, because the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible in the permanently defueled condition.
CHANNEL OPERATIONAL TEST (COT)	This definition is proposed for deletion, because the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible in the permanently defueled condition.

CORE ALTERATION	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term is no longer applicable since fuel will be permanently removed from the reactor core.
CORE OPERATING LIMITS REPORT (COLR)	This definition is proposed for deletion, because the term is not used in any PDTS specification. Technical Specification 5.6.5 that requires the COLR is also proposed for elimination.
DOSE EQUIVALENT I-131	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term is used in current TS 3.4.16 and TS 3.7.17 to express the specific activity limit from a mixture of iodine isotopes contained in reactor coolant and secondary coolant. Technical Specification 3.4.16 and TS 3.7.17 are proposed for deletion in the PDTS. The specific activity limit is used as the basis in accident analysis involving coolant releases. Since accident conditions associated with the RCS and secondary coolant system will no longer apply to the permanently shut down and defueled facility, the definition is no longer meaningful.
DOSE EQUIVALENT XE-133	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term is used in current TS 3.4.16 to express the specific activity limit from a mixture of xenon isotopes contained in reactor coolant. Technical Specification 3.4.16 is proposed for deletion in the PDTS. The specific activity limit is used as the basis in accident analysis involving coolant releases. Since accident conditions associated with the RCS and secondary coolant system will no longer apply to the permanently shut down and defueled facility, the definition is no longer meaningful.
L_a	This definition is proposed for deletion, because the term is not used in any PDTS. Technical Specification 5.5.15 that refers to L_a is also proposed for elimination. L_a is the maximum allowable primary containment leakage rate. Since accident conditions occurring inside the primary containment will no longer apply to the permanently shut down and defueled facility, the definition is no longer meaningful.

LEAKAGE	This definition is proposed for deletion, because the term is not used in any PDTS specification. Refer to the discussions for the proposed deletion of TS 3.4.13 and TS 5.5.8.
MASTER RELAY TEST	This definition is proposed for deletion, because the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible in the permanently defueled condition.
MODE	This definition, including Table 1.1-1, is proposed for deletion, because operational MODES are not used in any PDTS specification. MODES as defined in Table 1.1-1 are defined for operating or refueling conditions. This term does not apply to a facility in the permanently defueled condition.
OPERABLE – OPERABILITY	This definition is proposed for deletion, because the term is not used in any PDTS specification. There are no systems or components required to be operable in the PDTS, because there are no active systems, structures or components required to function to mitigate any of the remaining DBAs.
PHYSICS TESTS	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term does not apply to a facility in the permanently defueled condition.
QUADRANT POWER TILT RATIO (QPTR)	This definition is proposed for deletion, because the term is not used in any PDTS specification. This definition only applies to an operating reactor core.
RATED THERMAL POWER (RTP)	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
SHUTDOWN MARGIN (SDM)	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
SLAVE RELAY TEST	This definition is proposed for deletion, because the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible in the permanently defueled condition.

STAGGERED TEST BASIS	This definition is proposed for deletion, because the term is not used in any PDTS specification. This definition applies to the performance of surveillance tests on systems with multiple subsystems or channels. There are no surveillance requirements in the PDTS for operating systems.
THERMAL POWER	This definition is proposed for deletion, because the term is not used in any PDTS specification. This term is meaningful only to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	This definition is proposed for deletion, because the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accidents that remain credible in the permanently defueled condition.

TECHNICAL SPECIFICATION SECTION 1.2, LOGICAL CONNECTORS

Technical Specification 1.2, "Logical Connectors," explain the meaning of logical connectors. It is modified to reflect the logical connectors that continue to exist in the TSs.

<u>Current Purpose</u> Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies…	<u>Proposed Purpose</u> Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete ~~Conditions,~~ Required Actions~~, Completion Times,~~ ***and*** Surveillances~~, and Frequencies~~…
<u>Current Background</u> 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.	<u>Proposed Background</u> 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~~.
<u>Current Examples</u> EXAMPLES The following examples illustrate the use of logical connectors… Example 1.2-2…	<u>Proposed Example~~s~~</u> EXAMPLE~~S~~ The following example~~s~~ illustrate***s*** the use of logical connectors… Example 1.2-2 is proposed for deletion.

<u>Basis</u>

This section is modified to reflect the logical connectors utilized in TS 3.7.15. This is the only TS that utilizes logical connectors in the PDTS. These changes are administrative changes.

TECHNICAL SPECIFICATION SECTION 1.3, COMPLETION TIMES	
Technical Specification 1.3, "Completion Times," establishes the Completion Time convention and provides guidance for its use. It is modified to reflect the permanently shut down and defueled condition and the Completion Times that continue to exist in the PDTS.	
Current Background Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit...	Proposed Background Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~***handling and storage of spent nuclear fuel***...
Basis The Background section of TS 1.3 is modified to reflect the upcoming change in status regarding IP3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the primary mission will change from the safe operation of the unit to the safe handling and storage of spent nuclear fuel.	
Current 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 unit is in a MODE or specified condition stated in the Applicability of the LCO. Unless otherwise specified,... Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered...	Proposed 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 ~~unit~~***facility*** is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. ~~Unless otherwise specified...~~ Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the ~~unit~~**facility** is not within the LCO Applicability. ~~If situations are discovered...~~
Basis The Description section of TS 1.3 is modified to reflect the upcoming change in status regarding IP3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the PDTS will contain no operability requirements for any equipment. In addition, the term facility better represents IP3 in the permanently shut down and defueled condition.	

Current Examples	Proposed Example
EXAMPLES The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.	EXAMPLES~~S~~ The following examples~~s~~ illustrate***s*** the use of Completion Times with different ~~types of Conditions and changing Conditions~~***Required Actions***.
Example 1.3-1…	Example 1.3-1 is modified to address Completion Times as utilized by TS 3.7.15.
Example 1.3-2…	Example 1.3-2 is proposed for deletion.
Example 1.3-3…	Example 1.3-3 is proposed for deletion.
Example 1.3-4…	Example 1.3-4 is proposed for deletion.
Example 1.3-5…	Example 1.3-5 is proposed for deletion.
Example 1.3-6…	Example 1.3-6 is proposed for deletion.
Example 1.3-7…	Example 1.3-7 is proposed for deletion.

Basis

This section is modified to reflect the use of Completion Times that are utilized in TS 3.7.14, TS 3.7.15, and TS 3.7.16. These are the only TSs that have Completion Times in the PDTS. The changes to the Examples section of TS 1.3 are administrative changes.

TECHNICAL SPECIFICATION SECTION 1.4, FREQUENCY

Technical Specification 1.4, "Frequency," defines the proper use and application of Frequency requirements. It is modified to reflect the permanently shut down and defueled condition and the Frequencies that continue to exist in the PDTS.

Current Description	Proposed Description
…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.	…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~~.
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	~~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.	~~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.~~
<u>Current Examples</u> 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. Example 1.4-1... Example 1.4-2... Example 1.4-3...	<u>Proposed Example</u> EXAMPLE~~S~~ The following example~~s 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~~***illustrates the type of Frequency statement that appears in the Technical Specifications (TS)***. Example 1.4-1 is modified to address an example of a Frequency that is utilized by TS 3.7.14. Example 1.4-2 is proposed for deletion. Example 1.4-3 is proposed for deletion.

<u>Basis</u>

Technical Specification 1.4 is modified to reflect the upcoming change in status regarding IP3. This includes modifications to the description section, and to the examples. These proposed changes are administrative changes that reflect the changes to the other TSs and the remaining requirements.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the number and types of Surveillance Requirements that remain in the TSs are limited to those in TS 3.7.14, TS 3.7.15, and TS 3.7.16. This section is modified to provide the rules of usage and examples that continue to be applicable for those TSs.

Example 1.4-1 is modified to address an example of a Frequency that is utilized by TS 3.7.14. This includes the elimination of the references to the term "operational," inoperable equipment, Modes, Example 1.4-3, and LCO 3.0.4, and replacing the term "unit" with "facility." Examples 1.4-2 and 1.4-3 are eliminated.

TECHNICAL SPECFICATION SECTION 2.0, ~~SAFETY LIMITS (SLS)~~DELETED

Technical Specification 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 RCS pursuant to 10 CFR 50.36(c)(1).

Technical Specification Section 2.0 is proposed for deletion in its entirety, since the safety limits do not apply to a reactor that is in a permanently defueled condition.

A mark-up is provided to identify the section as deleted, because the TSs will not be renumbered.

Current TS 2.1 TS 2.0 SLs	Proposed TS 2.1 TS 2.0 ~~SLs~~***DELETED***
TS 2.1.1, Reactor Core SLs	Technical Specification 2.1.1 is proposed for deletion.
TS 2.1.2, RCS Pressure SL	Technical Specification 2.1.2 is proposed for deletion.
TS 2.2, SL Violations	Technical Specification 2.2 is proposed for deletion.
TS 2.2.1, "If SL 2.1.1 is violated..."	Technical Specification 2.2.1 is proposed for deletion.
TS 2.2.2, "If SL 2.1.2 is violated..."	Technical Specification 2.2.2 is proposed for deletion.

Basis

Technical Specifications 2.0, 2.1 and 2.2 are proposed for deletion in their entirety.

The restrictions of TS 2.1.1 prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. It is applicable in MODES 1 and 2. Since TS 2.1.1 applies to an operating reactor, its restrictions have no function in the permanently defueled condition.

The restriction of TS 2.1.2 protects the integrity of the RCS from over-pressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere. It is applicable in MODES 1 through 5, and MODE 6 when the reactor pressure vessel head is on. Since TS 2.1.2 applies to maintaining the RCS pressure, its restriction has no function in the permanently defueled condition.

Technical Specification 2.2.1 defines the action to take if SL 2.1.1 is not met. It requires the unit to be placed in MODE 3. It is deleted, because SL 2.1.1 is deleted.

TS 2.2:1 defines the action to take if SL 2.1.2 is not met. If the unit is in MODE 1 or 2, it requires the unit to be placed in MODE 3 within 1 hour. If the unit is MODE 3, 4, 5, or 6, it requires compliance to be restored within 5 minutes. It is deleted, because SL 2.1.2 is deleted.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). The safety limits and safety limit violations TSs apply to the reactor core and the RCS, they have no function in the permanently defueled condition. These specifications do not apply to the safe storage and handling of spent fuel in the SFP.

**TECHNICAL SPECIFICATION SECTION 3.0,
LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY**

Technical Specification Section 3.0 contains the general requirements applicable to all Limiting Conditions for Operation (LCOs) and applies at all times unless otherwise stated in a TS. Proposed revisions to these TSs (including those proposed for deletion) are described below. The corresponding TSs Bases are also being revised to reflect these changes.

A mark-up of this section is provided.

Current LCO 3.0.1	Proposed LCO 3.0.1
LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7 and LCO 3.0.8.	LCOs shall be met during the ~~MODES or other~~ specified conditions in the Applicability, except as provided in LCO 3.0.2~~, LCO 3.0.7 and LCO 3.0.8~~.

Basis

MODES as defined in Table 1.1-1 are defined for operating or refueling conditions. MODES are not used in any PDTS specification. Thus, the reference to MODES is deleted, because this term does not apply to a facility in the permanently defueled condition.

In addition, the references to LCOs 3.0.7 and 3.0.8 are deleted to reflect the proposed deletion of those LCOs discussed below.

Current LCO 3.0.2	Proposed 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…	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~~…

Basis

LCO 3.0.2 is modified by eliminating the references to LCOs 3.0.5 and 3.0.6. This change reflects the proposed deletion of those LCOs as discussed below.

LCO 3.0.3	This LCO is proposed for deletion.

Basis

LCO 3.0.3 provides the actions that must be implemented when an LCO is not met. It is only applicable in MODES 1 through 5. Pursuant to 10 CFR 50.82(a)(2), the facility license for IP3 will no longer authorize operation of the reactor or placement or retention of fuel in the reactor. Thus, references to operating MODES is no longer relevant. Thus, LCO 3.0.3 is no longer applicable in the permanently defueled condition.

LCO 3.0.4	This LCO is proposed for deletion.
Basis LCO 3.0.4 provides limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. LCO 3.0.4 is not proposed for inclusion in the PDTS since all actions in the remaining TS (i.e., TS 3.7.14, TS 3.7.15, and TS 3.7.16) have a completion time of "Immediately." This makes LCO 3.0.4 unnecessary. Thus, LCO 3.0.4 is no longer applicable in the permanently defueled condition.	
LCO 3.0.5	This LCO is proposed for deletion.
Basis LCO 3.0.5 provides the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The allowance of LCO 3.0.5 to not comply with the requirements of LCO 3.0.2 (i.e., to not comply with the Required Actions) to allow the performance of SRs on equipment declared inoperable or removed from service is no longer required. The remaining permanently defueled TSs ACTIONS do not include requirements to declare equipment inoperable or to remove it from service.	
LCO 3.0.6	This LCO is proposed for deletion.
Basis LCO 3.0.6 addresses the actions required for a supported system when the support system LCO is not met. It is proposed for deletion since there are no LCOs for equipment to be operable or in operation in the PDTS.	
LCO 3.0.7	This LCO is proposed for deletion.
Basis LCO 3.0.7 pertains to certain special tests and operations required to be performed at various times over the life of the unit. It is proposed for deletion since special tests and operations are not applicable to a permanently defueled facility.	
LCO 3.0.8	This LCO is proposed for deletion.
Basis LCO 3.0.8 addresses the actions required when one or more required snubbers are unable to perform their associated support function(s). It is proposed for deletion, because there are no LCOs for equipment to be operable or in operation in the PDTS. Thus, snubbers are not required to support any TS function.	

TECHNICAL SPECFICATION SECTION 3.0,
SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

Technical Specification Section 3.0 contains the general requirements applicable to all SRs and applies at all times unless otherwise stated in a TS. Proposed revisions to these TSs are described below. The corresponding TSs Bases are also being revised to reflect these changes.

A mark-up of this section is provided.

Current SR 3.0.1	Proposed 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...Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.	SRs shall be met during the ~~MODES or other~~ specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR...Surveillances do not have to be performed on ~~inoperable equipment or~~ variables outside specified limits.

Basis

SR 3.0.1 is modified by deleting the reference to MODES. Pursuant to 10 CFR 50.82(a)(2), the facility license for IP3 will no longer authorize operation of the reactor or placement or retention of fuel in the reactor.

MODES are not used in any PDTS specification. MODES as defined in Table 1.1-1 are for operating or refueling conditions. This term does not apply to a facility in the permanently defueled condition.

In addition, SR 3.0.1 is modified by eliminating the discussion regarding inoperable equipment. The remaining LCOs do not include any equipment operability requirements.

<table>
<tr><td><u>Current SR 3.0.2</u>

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.</td><td><u>Proposed SR 3.0.2</u>

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 <s>or as measured from the time a specified condition of the Frequency is met</s>.

<s>For Frequencies specified as "once," the above interval extension does not apply.</s>

<s>If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.</s>

<s>Exceptions to this Specification are stated in the individual Specifications.</s></td></tr>
<tr><td colspan="2"><u>Basis</u>

SR 3.0.2 provides an allowance for extending the frequency for performance of a SR to 1.25 times the interval specified in the frequency to facilitate scheduling or unforeseen problems that may prevent performance during normal intervals. It is proposed for revision to remove conditions for frequencies that do not exist in PDTS TSs.</td></tr>
<tr><td><u>Current SR 3.0.3</u>

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.</td><td><u>Proposed SR 3.0.3</u>

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. <s>The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed.</s> A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed</td></tr>
</table>

<table>
<tr><td colspan="2">

<u>Basis</u>

SR 3.0.3 provides an allowance to delay declaring an LCO not met, when a surveillance is not performed within its required frequency. This requirement is revised by removing language that was included within License Amendment No. 266 to the IP3 Facility Operating License to incorporate TSTF-529 (Reference 5). This language identified that the delay permitted by the specification was only applicable when there is a reasonable expectation that the surveillance will be met when performed. This requirement is not necessary in the permanently shut down and defueled condition, because SR 3.7.14.1, SR 3.7.15.1, and SR 3.7.16.1 are the only remaining surveillance requirements in the PDTS. These surveillances verify spent fuel pit water level, spent fuel pit boron concentration, and the initial enrichment and burnup of each fuel assembly and that the storage location meets LCO 3.7.16 requirements. These activities are not complex. Thus, there is a reasonable expectation that they will be met when performed.

</td></tr>
<tr><td>

<u>Current SR 3.0.4</u>

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 unit.

</td><td>

<u>Proposed SR 3.0.4</u>

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 unit.~~

</td></tr>
<tr><td colspan="2">

<u>Basis</u>

SR 3.0.4 is modified by deleting the reference to MODES. Pursuant to 10 CFR 50.82(a)(2), the facility license for IP3 will no longer authorize operation of the reactor or placement or retention of fuel in the reactor.

MODES are not used in any PDTS specification. MODES as defined in Table 1.1-1 are for operating or refueling conditions. This term does not apply to a facility in the permanently defueled condition.

In addition, SR 3.0.4 is modified by eliminating the provision that states that it shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The only remaining TSs with Required Actions are TS 3.7.14, TS 3.7.15, and TS 3.7.16, and they do not contain any Required Actions that would require an entry into another specified condition defined in the Applicability of a TS.

In addition, pursuant to 10 CFR 50.82(a)(2), the facility license for IP3 will no longer authorize operation of the reactor or placement or retention of fuel in the reactor. Thus, there will be no ACTIONS that require the shutdown of a unit.

</td></tr>
</table>

TECHNCIAL SPECIFICATION SECTION 3.1, REACTIVITY CONTROL SYSTEMS

Technical Specification Section 3.1 contains requirements to assure and verify operability of reactivity control systems.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). As a result, reactivity control systems will not be required and the requirements in TS Section 3.1 will not apply in the permanently defueled condition.

Technical Specification Section 3.1 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.1.1, SHUTDOWN MARGIN (SDM)	Technical Specification 3.1.1 is proposed for deletion. Technical Specification 3.1.1 is applicable in MODE 2 with k_{eff} < 1.0, and MODES 3 through 5. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in the applicable MODES and specified conditions will no longer occur. As a result, TS 3.1.1 will not apply in the permanently defueled condition.
TS 3.1.2, Core Reactivity	Technical Specification 3.1.2 is proposed for deletion. Technical Specification 3.1.2 is applicable in MODES 1 and 2. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in MODES 1 and 2 will no longer occur. As a result, TS 3.1.2 will not apply in the permanently defueled condition.

TS 3.1.3, Moderator Temperature Coefficient (MTC)	Technical Specification 3.1.3 is proposed for deletion. Technical Specification 3.1.3 is applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit and MODES 1, 2, and 3 for the lower MTC limit. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in the applicable MODES and specified conditions will no longer occur. As a result, this TS will not apply in the permanently defueled condition.
TS 3.1.4, Rod Group Alignment Limits	Technical Specification 3.1.4, including Table 3.1.4-1, is proposed for deletion. Technical Specification 3.1.4 is applicable in MODES 1 and 2. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in MODES 1 and 2 will no longer occur. As a result, this TS will not apply in the permanently defueled condition.
TS 3.1.5, Shutdown Bank Insertion Limits	Technical Specification 3.1.5 is proposed for deletion. Technical Specification 3.1.5 is applicable in MODE 1 and MODE 2 with any control bank not fully inserted. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in MODES 1 and 2 will no longer occur. As a result, this TS will not apply in the permanently defueled condition.

TS 3.1.6, Control Bank Insertion Limits	Technical Specification 3.1.6 is proposed for deletion. Technical Specification 3.1.6 is applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in the applicable MODES and specified conditions will no longer occur. As a result, this TS will not apply in the permanently defueled condition.
TS 3.1.7, Rod Position Indication	Technical Specification 3.1.7 is proposed for deletion. Technical Specification 3.1.7 is applicable in MODES 1 and 2. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in MODES 1 and 2 will no longer occur. As a result, this TS will not apply in the permanently defueled condition.
TS 3.1.8, PHYSICS TESTS Exceptions – MODE 2	Technical Specification 3.1.8 is proposed for deletion. Technical Specification 3.1.8 is applicable in MODE 2 during PHYSICS TESTS. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in the applicable MODE and specified condition will no longer occur. As a result, this TS will not apply in the permanently defueled condition.

TECHNICAL SPECIFICATION SECTION 3.2, POWER DISTRIBUTION LIMITS

Technical Specification Section 3.2 contains power distribution limits that provide assurance that fuel design criteria are not exceeded and the accident analysis assumptions remain valid.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). As a result, the requirements in TS Section 3.2 will not apply in the permanently defueled condition.

Technical Specification Section 3.2 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$)	Technical Specification 3.2.1 is proposed for deletion. Technical Specification 3.2.1 is applicable in MODE 1. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in MODE 1 will no longer occur. As a result, this TS will not apply in the permanently defueled condition.
TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	Technical Specification 3.2.2 is proposed for deletion. Technical Specification 3.2.2 is applicable in MODE 1. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in MODE 1 will no longer occur. As a result, this TS will not apply in the permanently defueled condition.

TS 3.2.3, AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)	Technical Specification 3.2.3 is proposed for deletion. Technical Specification 3.2.3 is applicable in MODE 1 with Thermal Power > 15% RTP. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in the applicable MODE and specified condition will no longer occur. As a result, this TS will not apply in the permanently defueled condition.
TS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)	Technical Specification 3.2.4 is proposed for deletion. Technical Specification 3.2.4 is applicable in MODE 1 with Thermal Power > 50% RTP. It will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, operation in the applicable MODE and specified condition will no longer occur. As a result, this TS will not apply in the permanently defueled condition.

TECHNICAL SPECIFICATION SECTION 3.3, INSTRUMENTATION

Technical Specification Section 3.3 contains operability requirements for sensing and control instrumentation required for safe operation of the facility.

After the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3, the 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel. The TSs that do not apply in the permanently defueled condition, or for structures, systems, or components that are not needed for accident mitigation in the defueled condition are being proposed for deletion.

Technical Specification Section 3.3 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.3.1, Reactor Protection System (RPS) Instrumentation	Technical Specification 3.3.1, including Table 3.3.1-1, is proposed for deletion. Dependent on function as defined in Table 3.3.1-1, TS 3.3.1 is applicable in various portions of MODES 1 through 5 or other specified conditions in those MODES.

	After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Operation in the applicable MODES and specified conditions will no longer occur. Thus, the RPS will not be required in the permanently defueled condition.
TS 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation	Technical Specification 3.3.2, including Table 3.3.2-1, is proposed for deletion. Dependent on function as defined in Table 3.3.2-1, TS 3.3.2 is applicable in various portions of MODES 1 through 4 or other specified conditions in those MODES. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in the applicable MODES and specified conditions will no longer occur. As a result, the ESFAS instrumentation will not be required in the permanently defueled condition.

TS 3.3.3, Post Accident Monitoring (PAM) Instrumentation	Technical Specification 3.3.3, including Table 3.3.3-1, is proposed for deletion. Technical Specification 3.3.3 is applicable in MODES 1, 2, and 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 will no longer occur. As a result, the PAM instrumentation will not be required in the permanently defueled condition.
TS 3.3.4, Remote Shutdown	Technical Specification 3.3.4 is proposed for deletion. Technical Specification 3.3.4 is applicable in MODES 1, 2, and 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 will no longer occur. As a result, the remote shutdown functions will not be required in the permanently defueled condition.
TS 3.3.5, Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	Technical Specification 3.3.5 is proposed for deletion Technical Specification 3.3.5 is applicable in MODES 1 through 4 and when the associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources – Shutdown." After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. In addition, TS 3.8.2 is proposed for deletion as discussed in the table for Section 3.8. The postulated DBAs and events associated with reactor or power operation analyzed in UFSAR

	Chapter 14 are no longer applicable in the permanently defueled condition. The analyses of the remaining DBAs (i.e., the FHA and the accidental release of waste liquid or waste gas) do not rely on Alternating Current (AC) electrical power sources for accident mitigation (dose consequences are acceptable without relying on any electrically-powered SSCs to remain functional during and following the event). There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of these events with the IP3 permanently shut down and defueled. As a result, the LOP DG start instrumentation will not be required in the permanently defueled condition.
TS 3.3.6, Containment Purge System and Pressure Relief Line Isolation Instrumentation	Technical Specification 3.3.6, including Table 3.3.6-1, is proposed for deletion. Technical Specification 3.3.6 is applicable in MODES 1 through 4 and during CORE ALTERATIONS, and during movement of irradiated fuel assemblies within containment. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 and CORE ALTERATIONS will no longer occur. In addition, the PDTS will not be implemented until after all of the fuel has been transferred from the reactor to the spent fuel pit. Thus, movement of irradiated fuel assemblies within the containment will no longer occur. As a result, the containment purge system and pressure relief line isolation instrumentation will not be required in the permanently defueled configuration.
TS 3.3.7, Control Room Ventilation System (CRVS) Actuation Instrumentation	Technical Specification 3.3.7, including Table 3.3.7-1, is proposed for deletion. Technical Specification 3.3.7 is applicable in MODES 1 through 4. After the certifications required by 10 CFR

	50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, CRVS actuation instrumentation will not be required in the permanently defueled configuration.
TS 3.3.8, Fuel Storage Building Emergency Ventilation System (FSBEVS) Actuation Instrumentation	Technical Specification 3.3.8 is proposed for deletion. Technical Specification 3.3.8 is applicable during movement of recently irradiated fuel in the fuel storage building. The Bases for TS 3.3.8 defines "recently irradiated" fuel as fuel that has occupied part of a critical reactor core within the previous 84 hours. The PDTS will not be implemented until after that time period, so that the specific condition of Applicability will no longer occur. As a result, the FSBEVS will not be required in the permanently defueled configuration.

TECHNICAL SPECIFICATION SECTION 3.4, REACTOR COOLANT SYSTEM (RCS)

Technical Specification Section 3.4 contains requirements that provide for appropriate control of process variables, design features, or operating restrictions needed for appropriate functional capability of RCS equipment required for safe operation of the facility.

After the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3, the 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel. The TSs that do not apply in a defueled condition, or for structures, systems, or components that are not needed for accident mitigation in the defueled condition, are being proposed for deletion.

Technical Specification Section 3.4 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	Technical Specification 3.4.1 is proposed for deletion. Technical Specification 3.4.1 is applicable in MODE 1, with the pressurizer pressure limit not being applicable at specifically defined periods. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the

	reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 1 will no longer occur. As a result, the RCS pressure, temperature, and low DNB limits are no longer applicable in the permanently defueled condition.
TS 3.4.2, RCS Minimum Temperature for Criticality	Technical Specification 3.4.2 is proposed for deletion. Technical Specification 3.4.2 is applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in the applicable MODES and specified condition will no longer occur. As a result, the RCS minimum temperature for criticality limit is no longer applicable in the permanently defueled condition.

TS 3.4.3, RCS Pressure and Temperature (P/T) Limits	Technical Specification 3.4.3, including Figures 3.4.3-1 and 3.4.3-2, is proposed for deletion. Technical Specification 3.4.3 is applicable at all times. If the requirements are not met in MODE 1, 2, 3, or 4, and the parameter(s) is not restored to within limit or the RCS is determined to not be acceptable for continued operation in accordance with Required Action A.1 or A.2, then the unit is required to be placed in MODE 3 and eventually MODE 5 with RCS pressure < 500 psig. If the requirements are not met at any time in other than MODE 1, 2, 3, or 4, action is required to restore the parameter(s) to within limit in accordance with Required Action C.1 and determine that the RCS is acceptable for continued operation prior to entering MODE 4 in accordance with Required Action C.2. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 5 will no longer occur. As a result, the RCS P/T limits are no longer applicable in the permanently defueled condition.
TS 3.4.4, RCS Loops – MODES 1 and 2	Technical Specification 3.4.4 is proposed for deletion. Technical Specification 3.4.4 is applicable in MODES 1 and 2. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 and 2 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.4.5, RCS Loops – MODE 3	Technical Specification 3.4.5 is proposed for deletion. Technical Specification 3.4.5 is applicable in MODE 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 3 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.6, RCS Loops – MODE 4	Technical Specification 3.4.6 is proposed for deletion. Technical Specification 3.4.6 is applicable in MODE 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.7, RCS Loops – MODE 5, Loops Filled	Technical Specification 3.4.7 is proposed for deletion. Technical Specification 3.4.7 is applicable in MODE 5 with the RCS loops filled. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 5 with the RCS loops filled will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.4.8, RCS Loops – MODE 5, Loops Not Filled	Technical Specification 3.4.8 is proposed for deletion. Technical Specification 3.4.8 is applicable in MODE 5 with the RCS loops not filled. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 5 with the RCS loops not filled will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.9, Pressurizer	Technical Specification 3.4.9 is proposed for deletion. Technical Specification 3.4.9 is applicable in MODES 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.10, Pressurizer Safety Valves	Technical Specification 3.4.10 is proposed for deletion. Technical Specification 3.4.10 is applicable in MODES 1 through 3 and MODE 4 with all RCS cold leg temperatures > 330°F. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs)	Technical Specification 3.4.11 is proposed for deletion. Technical Specification 3.4.11 is applicable in MODES 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.12, Low Temperature Overpressure Protection (LTOP)	Technical Specification 3.4.12, including Figures 3.4.12-1 through 3.4.12-3, is proposed for deletion. Technical Specification 3.4.12 is applicable whenever the Residual Heat Removal System (RHR) System is not isolated from the RCS, in MODE 4 when any RCS cold leg temperature is ≤ 330°F, MODE 5, and MODE 6 when the reactor vessel head is on. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in the applicable MODES 4 through 6 will no longer occur. In addition, LTOP was provided to protect the RCS from over-pressurization transients during shut down, in part, by providing a sufficient size RCS vent. In permanent shut down, the RCS is partially drained and adequately vented to prevent over-pressurization. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.4.13, RCS Operational LEAKAGE	Technical Specification 3.4.13 is proposed for deletion. Technical Specification 3.4.13 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage	Technical Specification 3.4.14 is proposed for deletion. Technical Specification 3.4.14 is applicable in MODES 1 through 3 and MODE 4 with an exception for leakage limits for valves in the RHR flow path when in, or during the transition to or from, the RHR mode of operation. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.15, RCS Leakage Detection Instrumentation	Technical Specification 3.4.15 is proposed for deletion. Technical Specification 3.4.15 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.4.16, RCS Specific Activity	Technical Specification 3.4.16 is proposed for deletion. Technical Specification 3.4.16 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.4.17, Steam Generator (SG) Tube Integrity	Technical Specification 3.4.17 is proposed for deletion. Technical Specification 3.4.17 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

**TECHNICAL SPECIFICATION SECTION 3.5,
EMERGENCY CORE COOLING SYSTEMS (ECCS)**

Technical Specification Section 3.5 contains requirements that provide for appropriate functional capability of ECCS equipment required for mitigation of DBAs or transients so as to protect the integrity of a fission product barrier.

After the certifications required under 10 CFR 50.82(a)(1) have been docketed for IP3, the 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel. The TSs that do not apply in a defueled condition, or for structures, systems, or components that are not needed for accident mitigation in the defueled condition, are being proposed for deletion.

Technical Specification Section 3.5 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.5.1, Accumulators	Technical Specification 3.5.1 is proposed for deletion. Technical Specification 3.5.1 is applicable in MODES 1 and 2 and MODE 3 with RCS pressure > 1000 psig. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.5.2, ECCS – Operating	Technical Specification 3.5.2 is proposed for deletion. Technical Specification 3.5.2 is applicable in MODES 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.5.3, ECCS – Shutdown	Technical Specification 3.5.3 is proposed for deletion. Technical Specification 3.5.3 is applicable in MODE 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.5.4, Refueling Water Storage Tank (RWST)	Technical Specification 3.5.4 is proposed for deletion. Technical Specification 3.5.4 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TECHNICAL SPECIFICATION SECTION 3.6, CONTAINMENT SYSTEMS

Technical Specification Section 3.6 contains requirements that assure the integrity of the containment, depressurization and cooling systems, and containment isolation valves.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). The TSs that do not apply in a defueled condition, or for structures, systems, or components that are not needed for accident mitigation in the defueled condition, are being proposed for deletion.

Technical Specification Section 3.6 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.6.1, Containment	Technical Specification 3.6.1 is proposed for deletion. Technical Specification 3.6.1 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.6.2, Containment Air Locks	Technical Specification 3.6.2 is proposed for deletion. Technical Specification 3.6.2 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.6.3, Containment Isolation Valves	Technical Specification 3.6.3 is proposed for deletion. Technical Specification 3.6.3 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.6.4, Containment Pressure	Technical Specification 3.6.4 is proposed for deletion. Technical Specification 3.6.4 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TS 3.6.5, Containment Air Temperature	Technical Specification 3.6.5 is proposed for deletion. Technical Specification 3.6.5 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.6.6, Containment Spray System and Containment Fan Cooler System	Technical Specification 3.6.6 is proposed for deletion. Technical Specification 3.6.6 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.6.7, Recirculation pH Control System	Technical Specification 3.6.7 is proposed for deletion. Technical Specification 3.6.7 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.6.8, Not Used	Technical Specification 3.6.8 is proposed for deletion. The deletion of this placeholder is an administrative change.

TS 3.6.9, Isolation Valve Seal Water (IVSW) System	Technical Specification 3.6.9 is proposed for deletion. Technical Specification 3.6.9 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.6.10, Weld Channel and Penetration Pressurization System (WC&PPS)	Technical Specification 3.6.10 is proposed for deletion. Technical Specification 3.6.10 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TECHNICAL SPECIFICATION SECTION 3.7,
~~PLANT SYSTEMS~~*SPENT FUEL PIT REQUIREMENTS*

Technical Specification Section 3.7 provides requirements for the appropriate functional capability of plant equipment required for safe operation of the facility, including the plant being in a defueled condition.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). The TSs that do not apply in a defueled condition, or for structures, systems, or components that are not needed for accident mitigation in the defueled condition, are being proposed for deletion.

Technical Specification 3.7 is retitled to reflect that the remaining TSs address SFP requirements.

Technical Specification 3.7.1 through TS 3.7.13 and TS 3.7.17 are proposed for deletion in their entirety. Thus, mark-ups of these TSs are not provided.

Technical Specification 3.7.14 provides the limit regarding the SFP water level. It will be retained in the PDTS, and modified to eliminate the reference to LCO 3.0.3.

Technical Specification 3.7.15 provides requirements regarding the SFP boron concentration. It will be retained in the PDTS, and modified to eliminate the reference to LCO 3.0.3. Technical Specification 3.7.16 provides the limits for storing fuel assemblies in the SFP. It will be retained in the PDTS, and modified to eliminate the reference to LCO 3.0.3. Mark-ups of TS 3.7.14, TS 3.7.15, and TS 3.7.16 are provided in Attachment 1 to this enclosure.	
Current IP3 TS	Basis for Change
TS 3.7, PLANT SYSTEMS	Proposed TS 3.7, ~~PLANT SYSTEMS~~***SPENT FUEL PIT REQUIREMENTS*** The TS section is proposed to be retitled to reflect that the remaining TSs in the section deal with SFP requirements in a permanently shut down and defueled facility. This is an administrative change.
TS 3.7.1, Main Steam Safety Valves (MSSVs)	Technical Specification 3.7.1, including Tables 3.7.1-1 and 3.7.1-2, is proposed for deletion. Technical Specification 3.7.1 is applicable in MODES 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.2, Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)	Technical Specification 3.7.2 is proposed for deletion. Technical Specification 3.7.2 is applicable in MODE 1 and MODES 2 and 3 except when all MSIVs are closed. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.3, Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MFRVs), Main Feedwater Inlet Isolation Valves (MFIIVs)	Technical Specification 3.7.3 is proposed for deletion.

and Main Feedwater (MF) Low Flow Bypass Valves	Technical Specification 3.7.3 is applicable in MODES 1 through 3 except when each main feedwater and bypass line is isolated by a closed and de-activated motor/air operated valve or isolated by a closed manual valve. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 3 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.4, Atmospheric Dump Valves (ADVs)	Technical Specification 3.7.4 is proposed for deletion. Technical Specification 3.7.4 is applicable in MODES 1 through 3 and MODE 4 when the steam generator is relied upon for heat removal. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.5, Auxiliary Feedwater (AFW) System	Technical Specification 3.7.5 is proposed for deletion. Technical Specification 3.7.5 is applicable in MODES 1 through 3 and MODE 4 when the steam generator is relied upon for heat removal. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.6, Condensate Storage Tank (CST)	Technical Specification 3.7.6 is proposed for deletion.

	Technical Specification 3.7.6 is applicable in MODES 1 through 3 and MODE 4 when the steam generator is relied upon for heat removal. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.7 City Water (CW)	Technical Specification 3.7.7 is proposed for deletion. Technical Specification 3.7.7 is applicable in MODES 1 through 3 and MODE 4 when the steam generator is relied upon for heat removal. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.8, Component Cooling Water (CCW) System	Technical Specification 3.7.8 is proposed for deletion. Technical Specification 3.7.8 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.9, Service Water System (SWS)	Technical Specification 3.7.9 is proposed for deletion. Technical Specification 3.7.9 is applicable in MODES 1 through 4. After the certifications required by 10 CFR

	50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.10, Ultimate Heat Sink (UHS)	Technical Specification 3.7.10 is proposed for deletion. Technical Specification 3.7.10 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.7.11, Control Room Ventilation System (CRVS)	Technical Specification 3.7.11 is proposed for deletion. Technical Specification 3.7.11 is applicable in MODES 1 through 4, and during movement of recently irradiated fuel assemblies. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. In addition, the Bases for TS 3.7.11 defines "recently irradiated" fuel as fuel that has occupied part of a critical reactor core within the previous 84 hours. The PDTS will not be implemented until after that time period, so that the specific condition of Applicability will no longer occur. As a result, the CRVS will not be required in the permanently defueled configuration.
TS 3.7.12, Control Room Air Conditioning System (CRACS)	Technical Specification 3.7.12 is proposed for deletion. Technical Specification 3.7.12 is applicable in MODES 1 through 4.

	After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur.
TS 3.7.13, Fuel Storage Building Emergency Ventilation System (FSBEVS)	Technical Specification 3.7.13 is proposed for deletion. Technical Specification 3.7.13 is applicable during movement of recently irradiated fuel assemblies in the fuel storage building. The Bases for TS 3.7.13 defines "recently irradiated" fuel as fuel that has occupied part of a critical reactor core within the previous 84 hours. The PDTS will not be implemented until after that time period, so that the specific condition of Applicability will no longer occur. As a result, the FSBEVS will not be required in the permanently defueled configuration.
TS 3.7.14, Spent Fuel Pit Water Level	Technical Specification 3.7.14 is retained in the PDTS. The title of TS Section 3.7 is administratively changed from addressing "plant systems" to addressing "spent fuel pit requirements" to comport with the remaining PDTS Section 3.7 LCOs. In addition, the NOTE in Required Action A.1 (LCO 3.0.3 is not applicable) is proposed to be deleted to conform to the deletion of TS LCO 3.0.3 as previously proposed.
TS 3.7.15, Spent Fuel Pit Boron Concentration	Technical Specification 3.7.15 is retained in the PDTS. The title of TS Section 3.7 is administratively changed from addressing "plant systems" to addressing "spent fuel pit requirements" to comport with the remaining PDTS Section 3.7 LCOs. In addition, the NOTE in Required Action A.1 (LCO 3.0.3 is not applicable) is proposed to be deleted to conform to the deletion of TS LCO 3.0.3 as previously proposed.

TS 3.7.16, Spent Fuel Assembly Storage	Technical Specification 3.7.16, including Figure 3.7.16-1, is retained in the PDTS. The title of TS Section 3.7 is administratively changed from addressing "plant systems" to addressing "spent fuel pit requirements" to comport with the remaining PDTS Section 3.7 LCOs. In addition, the NOTE in Required Action A.1 (LCO 3.0.3 is not applicable) is proposed to be deleted to conform to the deletion of TS LCO 3.0.3 as previously proposed.
TS 3.7.17, Secondary Specific Activity	Technical Specification 3.7.17 is proposed for deletion. Technical Specification 3.7.17 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.

TECHNICAL SPECIFICATION SECTION 3.8, ELECTRICAL POWER SYSTEMS

Technical Specification Section 3.8 contains operability requirements that provide for appropriate functional capability of plant electrical equipment required for safe operation of the facility, including the plant being in a defueled condition.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).

The DBAs and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA and the accidental releases of waste liquid or waste gas. There are no active systems credited as part of the initial conditions of these analyses or as part of the primary success path for mitigation of these events with IP3 permanently shut down and defueled.

Technical Specification Section 3.8 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.8.1, AC Sources – Operating	Technical Specification 3.8.1 is proposed for deletion. Technical Specification 3.8.1 is applicable in MODES 1 through 4.

	After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.8.2, AC Sources – Shutdown	Technical Specification 3.8.2 is proposed for deletion. Technical Specification 3.8.2 is applicable in MODES 5 and 6, and during movement of irradiated fuel assemblies. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 5 and 6 will no longer occur. In addition, the DBAs and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA and the accidental releases of waste liquid or waste gas. There are no active systems credited as part of the initial conditions of these analyses or as part of the primary success path for mitigation of these events with IP3 permanently shut down and defueled. As a result, this TS will not be applicable in the permanently defueled configuration.
TS 3.8.3, Diesel Fuel Oil and Starting Air	Technical Specification 3.8.3 is proposed for deletion. Technical Specification 3.8.3 is applicable when the associated Diesel Generator (DG) is required to be OPERABLE. Technical Specification 3.8.1 and TS 3.8.2 provide the OPERABILITY requirements regarding the DGs. As previously discussed, these TSs are proposed for deletion. Thus, TS 3.8.3 is not included in the PDTS because the TSs that it supports are no longer required after IP3 is permanently shut down and defueled.

TS 3.8.4, DC Sources – Operating	Technical Specification 3.8.4 is proposed for deletion. Technical Specification 3.8.4 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.8.5, DC Sources – Shutdown	Technical Specification 3.8.5 is proposed for deletion. Technical Specification 3.8.5 is applicable in MODES 5 and 6, and during movement of irradiated fuel assemblies. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 5 and 6 will no longer occur. In addition, the DBAs and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA and the accidental releases of waste liquid or waste gas. There are no active systems credited as part of the initial conditions of these analyses or as part of the primary success path for mitigation of these events with IP3 permanently shut down and defueled. As a result, this TS will not be applicable in the permanently defueled configuration.
TS 3.8.6, Battery Cell Parameters	Technical Specification 3.8.6, including Table 3.8.6-1, is proposed for deletion. Technical Specification 3.8.6 is applicable when the associated Direct Current (DC) electrical power subsystems are required to be OPERABLE. Technical Specification 3.8.4 and TS 3.8.5 provide the OPERABILITY requirements regarding the DC

	sources. As previously discussed, these TSs are proposed for deletion. Thus, TS 3.8.6 is not included in the PDTS because the TSs that it supports are no longer required after IP3 is permanently shut down and defueled.
TS 3.8.7, Inverters – Operating	Technical Specification 3.8.7 is proposed for deletion. Technical Specification 3.8.7 is applicable in MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.8.8, Inverters – Shutdown	Technical Specification 3.8.8 is proposed for deletion. Technical Specification 3.8.8 is applicable in MODES 5 and 6, and during movement of irradiated fuel assemblies. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 5 and 6 will no longer occur. In addition, the DBAs and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA and the accidental releases of waste liquid or waste gas. There are no active systems credited as part of the initial conditions of these analyses or as part of the primary success path for mitigation of these events with IP3 permanently shut down and defueled. As a result, this TS will not be applicable in the permanently defueled configuration.
TS 3.8.9, Distribution Systems – Operating	Technical Specification 3.8.9 is proposed for deletion. Technical Specification 3.8.9 is applicable in

	MODES 1 through 4. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 1 through 4 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.8.10, Distribution Systems – Shutdown	Technical Specification 3.8.10 is proposed for deletion. Technical Specification 3.8.10 is applicable in MODES 5 and 6, and during movement of irradiated fuel assemblies. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODES 5 and 6 will no longer occur. In addition, the DBAs and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA and the accidental releases of waste liquid or waste gas. There are no active systems credited as part of the initial conditions of these analyses or as part of the primary success path for mitigation of these events with IP3 permanently shut down and defueled. As a result, this TS will not be applicable in a permanently defueled configuration.

TECHNICAL SPECIFICATION SECTION 3.9, REFUELING OPERATIONS

Technical Specification Section 3.9 contains requirements that provide for appropriate functional capability of parameters and equipment that are required for mitigation of DBAs during refueling operations (moving irradiated fuel to or from the reactor core).

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).

The DBAs and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA and the accidental releases of waste liquid or waste gas. There are no active systems credited as part of the initial conditions of these analyses or as part of the primary success path for mitigation of these events with IP3 permanently shut down and defueled.

Technical Specification Section 3.9 is proposed for deletion in its entirety. Thus, a mark-up of this TS section is not provided.

Current IP3 TS	Basis for Change
TS 3.9.1, Boron Concentration	Technical Specification 3.9.1 is proposed for deletion. Technical Specification 3.9.1 is applicable in MODE 6. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 6 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.9.2, Nuclear Instrumentation	Technical Specification 3.9.2 is proposed for deletion. Technical Specification 3.9.2 is applicable in MODE 6. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 6 will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.9.3, Containment Penetrations	Technical Specification 3.9.3 is proposed for deletion. Technical Specification 3.9.3 is applicable during movement of recently irradiated fuel assemblies within containment. As stated in the Bases for TS 3.9.3, the release of radioactivity from the containment following an FHA is limited by several conditions, including a minimum decay time of 84 hours prior to moving irradiated fuel. "Recently irradiated" fuel is fuel that has occupied part of a critical reactor core within the previous 84 hours. The PDTS will not be implemented until after that time period, so that the specific condition of Applicability will no longer occur.

	As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.9.4, Residual Heat Removal (RHR) and Coolant Circulation – High Water Level	Technical Specification 3.9.4 is proposed for deletion. Technical Specification 3.9.4 is applicable in MODE 6 with the water level ≥ 23 feet above the top of the reactor vessel flange. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 6 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.9.5, Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level	Technical Specification 3.9.5 is proposed for deletion. Technical Specification 3.9.5 is applicable in MODE 6 with the water level < 23 feet above the top of the reactor vessel flange. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, operation in MODE 6 with the specified condition will no longer occur. As a result, this TS will not be applicable in a permanently defueled condition.
TS 3.9.6, Refueling Cavity Water Level	Technical Specification 3.9.6 is proposed for deletion. Technical Specification 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and during movement of irradiated fuel assemblies within containment. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Thus, CORE ALTERATIONS with the

	specified conditions will no longer occur. In addition, the PDTS will not be implemented until after all of the fuel has been transferred from the reactor to the spent fuel pit. Thus, movement of irradiated fuel assemblies within the containment will no longer occur. As a result, this TS will no longer be applicable in the permanently defueled condition.

TECHNICAL SPECIFICATION SECTION 4.0, DESIGN FEATURES

Currently, TS Section 4.0, Design Features, provides information and design requirements associated with plant systems.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). The TSs that do not apply in a defueled condition, or for structures, systems, or components that are not needed for accident mitigation in the defueled condition, are being proposed for deletion.

Technical Specification 4.3.1.1 is modified by making grammatical corrections regarding punctuation.

Technical Specification 4.3.1.2 is proposed for deletion.

Current IP3 TS	Basis for Change
Current TS 4.3.1.1	Proposed TS 4.3.1.1
4.3.1.1 The spent fuel storage racks are designed and shall be maintained with: a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent; b. $k_{eff} \leq 0.95$ if assemblies are inserted in accordance with Technical Specification 3.7.16, Spent Fuel Assembly Storage. c. A nominal 9.075 inch center to center distance between fuel assemblies placed in the high density fuel storage racks (Region II); d. A nominal 10.76 inch center to center distance between fuel assemblies placed in low density fuel storage racks (Region I);	4.3.1.1 The spent fuel storage racks are designed and shall be maintained with: a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent; b. $k_{eff} \leq 0.95$ if assemblies are inserted in accordance with Technical Specification 3.7.16, Spent Fuel Assembly Storage~~.~~; c. A nominal 9.075 inch center to center distance between fuel assemblies placed in the high density fuel storage racks (Region II); d. A nominal 10.76 inch center to center distance between fuel assemblies placed in low density fuel storage racks (Region I)~~;~~.

Basis
Technical Specification 4.3.1.1 is modified to correct grammatical errors regarding punctuation.

Current TS 4.3.1.2	Proposed TS 4.3.1.2
The new fuel storage racks are designed and shall be maintained with...	None

Basis
Technical Specification 4.3.1.2 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Indian Point Nuclear Generating Station Unit 3 will never acquire new fuel again. Thus, this TS is not applicable in the permanently shut down and defueled condition.

TECHNICAL SPECIFICATION SECTION 5.0, ADMINISTRATIVE CONTROLS
TS Section 5.0 establishes the requirements associated with staffing, training, procedures, programs and reporting requirements. This section is proposed to be revised to include only those administrative requirements needed for safe storage and movement of fuel in the SFP. The UFSAR is proposed to be retitled as the DSAR and the references to UFSAR in Section 5.0 are replaced with DSAR. The DSAR is the document that will be maintained in accordance with 10 CFR 50.59 and 10 CFR 50.71(e) and remain applicable to IP3 in the permanently shut down and defueled condition.

Current TS 5.2.1	Proposed TS 5.2.1
... a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all decommissioning organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the facility specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the UFSAR and Quality Assurance Plan, as appropriate; ...	... a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all decommissioning organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the facility specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the ~~UFSAR~~***DSAR*** and Quality Assurance Plan, as appropriate; ...

Basis

Technical Specification 5.2.1 will be retained, but modified by replacing the reference to the UFSAR in TS 5.2.1.a with a reference to the DSAR.

Following the permanent shut down and defueling of IP3, the IP3 UFSAR will be updated to reflect this condition. The document will be retitled as the DSAR. Thus, this proposed change is an administrative change.

Current TS 5.4.1	Proposed TS 5.4.1
a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 except as provided in the quality assurance program described or referenced in the Updated FSAR. b. Deleted c. Quality assurance for effluent and environmental monitoring; d. Fire Protection Program implementation; and...	a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 except as provided in the quality assurance program described or referenced in the ~~Updated FSAR.~~***DSAR***; b. Deleted; c. Quality assurance for effluent and environmental monitoring; d. ~~Fire Protection Program implementation~~ ***Deleted***; and...

Basis

Following the permanent shut down and defueling of IP3, the IP3 UFSAR will be updated to reflect this condition. The document will be retitled as the DSAR. Thus, the proposed change to TS 5.4.1.a is an administrative change.

Technical Specification 5.4.1.d is proposed for deletion. This is consistent with the proposed deletion of License Condition 2.H. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the fire protection program will be revised to take into account the decommissioning facility conditions and activities. Indian Point Nuclear Generating Station Unit 3 will continue to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether the TSs contain a requirement to establish, implement, and maintain procedures for a fire protection program. Therefore, a TS requirement for fire protection program procedures for a permanently shut down and defueled facility is not needed.

TS 5.5.2, Primary Coolant Sources Outside Containment	The title for TS 5.5.2 is deleted. This is an administrative change, because the TS was previously deleted in another license amendment.
TS 5.5.4, Radioactive Effluent Controls Program	Technical Specification 5.5.4 will be retained, but TS 5.5.4.d, TS 5.5.4.h, and TS 5.5.4.i are proposed to be modified to replace "unit" with "unit/facility." These are administrative changes to the IP3 TSs. It establishes consistency with the IP2 PDTS.
TS 5.5.5, Component Cyclic or Transient Limit	Technical Specification 5.5.5 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Operation in MODES 1 through 6 will never occur again. Thus, this TS is not applicable in the permanently shut down and defueled condition.
TS 5.5.6, Reactor Coolant Pump Flywheel Inspection Program	Technical Specification 5.5.6 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, the reactor coolant pumps will no longer perform a function in the permanently shut down and defueled state. Technical Specification 5.5.6 is proposed for deletion to be consistent with the proposed deletion of TS 3.4.4 through TS 3.4.8. These TSs provide the operability requirements for the RCS loops. Given their proposed deletion, there is no need to maintain this support program.
TS 5.5.7, Inservice Testing Program	Technical Specification 5.5.7 is proposed for deletion. Technical Specification 5.5.7 provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. In the permanently shut down and defueled condition, there are no longer any ASME Code class pumps and valves that remain in operation and are relied upon to mitigate a DBA.

	As such, the inservice testing program will no longer be relevant in the permanently shut down and defueled condition.
TS 5.5.8, Steam Generator (SG) Program	Technical Specification 5.5.8 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the SG will no longer perform a function in the permanently shut down and defueled state. Technical Specification 3.4.17 provides the requirements to ensure SG tube integrity in MODES 1 through 4. It is proposed for deletion. Thus, the proposed deletion of this supporting TS program is appropriate.
TS 5.5.9, Secondary Water Chemistry Program	Technical Specification 5.5.9 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, there will be no need to monitor secondary water chemistry to inhibit steam generator tube degradation in the permanently defueled condition. Thus, the deletion of this TS is appropriate.
TS 5.5.10, Ventilation Filter Testing Program (VFTP)	Technical Specification 5.5.10 is proposed for deletion. As previously discussed, TS 3.6.6, Containment Spray System and Containment Fan Cooler System, and TS 3.7.11, Control Room Ventilation System (CRVS), are proposed for deletion. Thus, this support program is not required in the permanently shut down and defueled condition.

TS 5.5.12, Diesel Fuel Oil Testing Program	Technical Specification 5.5.12 is proposed for deletion. As previously discussed, TS 3.8.1, TS 3.8.2, and TS 3.8.3 are proposed for deletion. These TSs define the operability requirements regarding the diesel generators. Thus, this support program is not required in the permanently shut down and defueled condition.
TS 5.5.13, Technical Specifications (TS) Bases Control Program	Technical Specification 5.5.13 will be retained, but modified by replacing the references to the "updated UFSAR" and UFSAR in TS 5.5.13.b.2 and TS 5.5.13.c with references to the DSAR. Following the permanent shut down and defueling of IP3, the IP3 UFSAR will be updated to reflect this condition. The document will be retitled as the DSAR. Thus, these proposed changes are administrative changes.
TS 5.5.14, Safety Function Determination Program (SFDP)	Technical Specification 5.5.14 is proposed for deletion. This program was established to ensure loss of safety function is detected and appropriate actions taken. The LCOs remaining in the PDTS do not rely on the operability of any active equipment or systems to satisfy the LCO. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, there is no longer a need for redundant systems. Therefore, the requirements of the SFDP, which directs cross train checks of multiple and redundant safety systems, no longer apply. Additionally, the SFDP is invoked in LCO 3.0.6, which is being deleted in its entirety as previously discussed. Thus, the SFDP is not needed in a permanently shut down and defueled condition.
TS 5.5.15, Containment Leakage Rate Testing Program	Technical Specification 5.5.15 is proposed for deletion. The IP3 10 CFR 50 facility license will no longer authorize use of the facility for power operation or emplacement or retention of fuel in the reactor vessel as provided in 10 CFR Part 50.82(a)(2). Containment integrity is not credited in the analysis of the accidents that remain credible in the permanently defueled condition. In addition, TS 3.6.1 through TS 3.6.10 regarding the

	containment systems are proposed for deletion. Thus, the deletion of this TS is appropriate.
TS 5.5.16, Control Room Envelope Habitability Program	Technical Specification 5.5.16 is proposed for deletion. As previously discussed, TS 3.7.11, Control Room Ventilation System (CRVS), is proposed for deletion. Thus, this support program is not required in the permanently shut down and defueled condition.
TS 5.6.4, Not Used	The placeholder for TS 5.6.4 is proposed for deletion. This is an administrative change to reflect reorganization of the TS.
TS 5.6.5, Core Operating Limits Report (COLR)	Technical Specification 5.6.5 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, there will no longer be a need to establish core operating limits. As a result, this TS will not be applicable in the permanently defueled condition.
TS 5.6.6, Not Used	The placeholder for TS 5.6.6 is proposed for deletion. This is an administrative change to reflect reorganization of the TS.
TS 5.6.7, Post Accident Monitoring Instrumentation (PAM) Report	Technical Specification 5.6.7 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Technical Specification 3.3.3 provides the operability requirements for the PAM instrumentation. It is proposed for deletion. Given that the reporting requirements in Conditions B and F of LCO 3.3.3 are proposed for deletion, the proposed deletion of the TS 5.6.7 reporting details is appropriate.

TS 5.6.8, Steam Generator Tube Inspection Report	Technical Specification 5.6.8 is proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the SG will no longer perform a function in the permanently shut down and defueled state. Technical Specification 3.4.17 provides the requirements to ensure SG tube integrity in MODES 1 through 4. It is proposed for deletion. In addition, TS 5.5.8, Steam Generator (SG) Program, is proposed for deletion. Thus, the proposed deletion of this supporting TS program is appropriate.

APPENDIX B TO FACILITY ~~OPERATING~~ LICENSE	
<u>Current Cover Page for Part I</u> APPENDIX B TO FACILITY OPERATING LICENSE...	<u>Proposed Cover Page for Part I</u> APPENDIX B TO FACILITY ~~OPERATING~~ LICENSE...
<u>Basis</u> Appendix B, Part I, is modified by replacing the reference to "Facility Operating License" with a reference to "Facility License." This change reflects the upcoming change in status regarding IP3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	
<u>Current Section 1.0</u> 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.	<u>Proposed Section 1.0</u> The Environmental Protection Plan (EPP) is to provide for protection of environmental values during ~~construction and operation~~***handling and storage of spent fuel and maintenance*** of the nuclear facility. The principal objectives of the EPP are as follows: (1) Verify that the ~~plant is operated~~***facility is maintained*** 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 SPDES permit.	(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 ***handling and storage of spent fuel and maintenance of the*** 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 SPDES permit.
Basis The proposed changes to Section 1.0 replace a reference to "construction and operation" with a reference to "handling and storage of spent fuel and maintenance" and a reference to "plant is operated" with "facility is maintained," and a reference to "facility construction and operation" with "handling and storage of spent fuel and maintenance of the facility." These proposed changes reflect the revised mission of the facility in the permanently shut down and defueled condition.	
Current Section 3.1 ENO may make changes in station design or operations or perform tests or experiments...Changes in plant design or operation or performance of tests or experiments... A proposed change, test, or experiment shall...(2) a significant change in effluents or power level;...	Proposed Section 3.1 ENO may make changes in ~~station~~***facility*** design or operations or perform tests or experiments...Changes in ~~plant~~***facility*** design or operation or performance of tests or experiments... A proposed change, test, or experiment shall...(2) a significant change in effluents ~~or power level~~;...
Basis The proposed changes to Section 3.1 replace references to "station" and "plant" with references to "facility." These proposed changes reflect the revised mission of the facility in the permanently shut down and defueled condition. The proposed change to Section 3.1 to eliminate the reference to "power level" reflects the permanently shut down and defueled condition. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

<table>
<tr><td><u>Current Section 3.3</u>

Changes in plant design or operation and...</td><td><u>Proposed Section 3.3</u>

Changes in ~~plant~~facility design or operation and...</td></tr>
<tr><td colspan="2"><u>Basis</u>

The proposed change to Section 3.3 replaces the reference to "plant" with a reference to "facility." This proposed change reflects the revised mission of the facility in the permanently shut down and defueled condition.</td></tr>
<tr><td><u>Current Section 4.1</u>

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...</td><td><u>Proposed Section 4.1</u>

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to ~~plant operation~~the handling and storage of spent fuel and maintenance of the facility shall be recorded and...</td></tr>
<tr><td colspan="2"><u>Basis</u>

The proposed change to Section 4.1 replaces the reference to "plant operation" with a reference to "the handling and storage of spent fuel and maintenance of the facility." This proposed change reflects the revised mission of the facility in the permanently shut down and defueled condition.</td></tr>
<tr><td><u>Current Section 4.2</u>

...The currently applicable Biological Opinion concludes that continued operation of IP2 and IP3 is not likely to jeopardize the continued existence of the listed species or to adversely affect the designated critical habitat of those species.</td><td><u>Proposed Section 4.2</u>

...The currently applicable Biological Opinion concludes that continued operation of IP2 and IP3 is not likely to jeopardize the continued existence of the listed species or to adversely affect the designated critical habitat of those species. This Biological Opinion conservatively bounds the conditions that will occur in the permanently shut down and defueled condition.</td></tr>
<tr><td colspan="2"><u>Basis</u>

The proposed change to Section 4.2 concludes that the Biological Opinion rendered during the evaluation of the continued operation of IP2 and IP3 conservatively bounds the conditions that will occur in the permanently shut down and defueled condition. This addition clarifies that the permanent shut down and defueling of IP3 will not impact the Biological Opinion regarding the shortnose sturgeon and Atlantic sturgeon in an adverse manner when compared to the continued operation of IP2 and IP3.</td></tr>
</table>

<table>
<tr><td><u>Current Section 5.2</u>

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 the NRC on request.

Records of modifications to plant structures, systems and components determined to potentially affect the continued protection of the environmental 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.</td><td><u>Proposed Section 5.2</u>

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

Records of modifications to <s>plant</s>facility structures, systems and components determined to potentially affect the continued protection of the environment<s>al</s> shall be retained for the life of the <s>plant</s>facility. 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.</td></tr>
<tr><td colspan="2"><u>Basis</u>

The proposed changes to Section 5.2 clarify that the reference to "plant operation" refers to plant operations previous to the permanent shut down, and includes a reference to "the handling and storage of spent fuel and maintenance of the facility." In addition, references to "plant" are replaced with "facility," and an editorial correction to replace "environmental" with "environment" is also made. These proposed changes reflect the revised mission of the facility in the permanently shut down and defueled condition.</td></tr>
</table>

<table>
<tr><td><u>Current Section 5.4.1</u>

...and an assessment of the observed impacts of the plant operation on the environment...

(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...</td><td><u>Proposed Section 5.4.1</u>

...and an assessment of the observed impacts of the previous plant operation and the handling and storage of spent fuel and maintenance of the facility on the environment...

(b) A list of all changes in <s>station</s>facility design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental issue...</td></tr>
<tr><td colspan="2"><u>Basis</u>

The proposed changes to Section 5.4 clarify that the reference to "plant operation" refers to plant operations previous to the permanent shut down, and includes a reference to "the handling and storage of spent fuel and maintenance of the facility." In addition, a reference to "station" is replaced with a reference to "facility." These proposed changes reflect the revised mission of the facility in the permanently shut down and defueled condition.</td></tr>
</table>

Current Section 5.4.2	Proposed Section 5.4.2
...The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b)...	...The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and ~~plant operating characteristics~~***facility conditions***, (b)...
Basis The proposed change to Section 5.4.2 replaces a reference to "plant operating characteristics" with "facility conditions." This proposed change reflects the revised mission of the facility in the permanently shut down and defueled condition.	

APPENDIX B TO FACILITY ~~OPERATING~~ LICENSE	
Current Cover Page for Part II APPENDIX B TO FACILITY OPERATING LICENSE...	Proposed Cover Page for Part II APPENDIX B TO FACILITY ~~OPERATING~~ LICENSE...
Basis Appendix B, Part II, is modified by replacing the references to "Facility Operating License" with "Facility License." This change reflects the upcoming change in status regarding IP3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

APPENDIX C TO FACILITY ~~OPERATING~~ LICENSE	
Current Cover Page for Part I APPENDIX C TO FACILITY OPERATING LICENSE...	Proposed Cover Page for Part I APPENDIX C TO FACILITY ~~OPERATING~~ LICENSE...
Current Header for Part I Facility Operating License...	Proposed Header for Part I Facility ~~Operating~~ License...
Current Cover Page for Part II APPENDIX C TO FACILITY OPERATING LICENSE...	Proposed Cover Page for Part II APPENDIX C TO FACILITY ~~OPERATING~~ LICENSE...
Basis Appendix C, Parts I and II, are modified by replacing the references to "Facility Operating License" with "Facility License." These changes reflect the upcoming change in status regarding IP3. After the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

The mark-ups of the Appendix A TSs Bases and re-typed versions of the PDTS Bases are provided for information only. Upon approval of this amendment, changes to the Appendix A TSs Bases will be incorporated in accordance with TS 5.5.13, "Technical Specifications (TS) Bases Control Program."

3.1 APPLICABLE REGULATORY REQUIREMENT/CRITERIA

<u>10 CFR 50.82, Termination of License</u>

The 10 CFR 50.82(a)(1) paragraph requires that when a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of 10 CFR 50.4(b)(8), and once fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of 10 CFR 50.4(b)(9). On February 8, 2017, Entergy notified the NRC that IP3 would permanently cease operations no later than April 30, 2021 (Reference 1). Entergy recognizes that approval of these proposed changes is contingent upon the submittal of the certifications required by 10 CFR 50.82(a)(1).

The 10 CFR 50.82(a)(2) paragraph states: "Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel."

<u>10 CFR 50.36, Technical Specifications</u>

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TSs "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." (Statement of Consideration, "Technical Specification for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968))

Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a facilities' TSs.

These criteria, which were subsequently codified in changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953), also pertain to the TS requirements for safe storage of spent fuel. A general discussion of these considerations is provided below to address the existing LCOs.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for "installed instrumentation that is used to detect, and indicate in the Control Room, a significant abnormal degradation of the reactor coolant pressure boundary." Since no fuel will be present in the reactor or RCS at IP3 in the permanently shut down and defueled condition, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some DBAs which continue to apply to a facility authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a facility with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The applicable DBAs for IP3 in the permanently defueled condition are discussed in more detail within this license amendment request.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The intent of this criterion is to capture into the TSs only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that will continue to apply to IP3, there are still DBAs that will continue to apply to a facility authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a facility with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The scope of DBAs that will be applicable to IP3 is discussed in more detail within this license amendment request.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs that operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. All of the accident sequences that previously dominated risk at IP3 will no longer be applicable after the reactor is in the permanently shut down and defueled condition.

Addressing administrative controls, 10 CFR 50.36(c)(5) states that they"...are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." This license amendment request is proposing changes to the Administrative Controls section, with conforming changes proposed to additional sections, consistent with the pending decommissioning status of the plant. This request applies the principles identified in 10 CFR 50.36(c)(6), Decommissioning, for a facility which has submitted certifications required by 50.82(a)(1) and proposes changes to the Administrative Controls appropriate for the IP3 permanently defueled condition. As 10 CFR 50.36(c)(6) states, this type of change should be considered on a case-by-case basis.

The 10 CFR 50.36(c)(6), "Decommissioning," provisions apply only to nuclear power reactor facilities that have submitted the certifications required by 10 CFR 50.82(a)(1). For such facilities, TSs involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

This proposed amendment deletes the portions of the previous IP3 TSs that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shut down and defueled condition.

10 CFR 50.48(f), Fire Protection During Decommissioning

The 10 CFR 50.48(f) paragraph states, in part, that: "Licensees that have submitted the certifications required under 10 CFR 50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (i.e., that could result in a radiological hazard)...

(1) The objectives of the fire protection program are to —

    (i) Reasonably prevent these fires from occurring;
    (ii) Rapidly detect, control, and extinguish those fires that do occur and that could result in a radiological hazard; and
    (iii) Ensure that the risk of fire-induced radiological hazards to the public environment and plant personnel is minimized.

(2) The licensee shall assess the fire protection program on a regular basis. The licensee shall revise the plan as appropriate throughout the various stages of facility decommissioning.

(3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities."

10 CFR 50.51, Continuation of License

The 10 CFR 50.51(b) paragraph 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. During such period of continued effectiveness, the licensee shall:

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility."

10 CFR 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants

Section 1.3 of the IP3 UFSAR states:

"The General Design Criteria presented and discussed in specific sections of the FSAR (which describe systems, structures, equipment and components important to safety) are those which were in effect at the time when Indian Point 3 was designed and constructed. The General Design Criteria which formed the bases for the Indian Point 3 design were published by the Atomic Energy Commission in the Federal Register of July 11, 1967 and subsequently made part of 10 CFR 50.

The Authority completed a study of the method by which the Indian Point 3 facility complied with the safety rules and regulations, in particular those contained in 10 CFR Parts 20 and 50, that were in effect at the time of the study. The study was conducted in accordance with the provisions of NRC Confirmatory Order of February 11, 1980 and were submitted to the NRC on August 11, 1980. The NRC audit of submittal indicated that the Indian Point 3 design and operation meet the applicable regulations. The following sections provide the results of the compliance study, updated to reflect changes made to the configuration since the study was completed."

The Indian Point Nuclear Generating Station Unit 3 design and licensing basis for fuel storage and handling and radiological controls is detailed in the UFSAR and other plant-specific licensing basis documents.

10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

The 10 CFR 50.46(a)(1)(i) paragraph states: "This section does not apply to a nuclear power reactor facility for which the certifications required under 10 CFR 50.82(a)(1) have been submitted."

10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants

The 10 CFR 50.62(a) paragraph states: "The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted."

Design Basis Accidents (DBAs)

Section 14 of the IP3 UFSAR describes the DBA scenarios that are applicable during plant operations. After certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and they are docketed for IP3, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). With the reactor in a permanently shut down and defueled condition, the facility mission changes. The primary mission is now the safe storage and handling of irradiated fuel. In this condition, the spectrum of credible accidents is much smaller than for an operational plant. Therefore, most of the accident scenarios postulated in UFSAR Section 14 will no longer be applicable after IP3 is in the permanently defueled condition. The only remaining DBAs will be the FHA and the accidental release of waste liquid or waste gas. This license amendment request includes additional discussion regarding the analyses of these accidents.

3.2 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.92, Entergy Nuclear Operations, Inc. (Entergy) has reviewed the proposed changes and concludes that the changes do not involve a significant hazards consideration since the proposed changes satisfy the criteria in 10 CFR 50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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.

On February 8, 2017, Entergy notified the U.S. Nuclear Regulatory Commission (NRC) that it would permanently cease power operations at Indian Point Nuclear Generating Station Unit No. 3 (IP3) no later than April 30, 2021 (Reference 1). After the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed for IP3, the 10 CFR Part 50 license for IP3 will no longer authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2).

This proposed license amendment would: revise the IP3 Facility Operating License (FOL); revise the Technical Specifications (TSs) in Appendix A of the FOL to Permanently Defueled Technical Specifications (PDTS); revise the Environmental Technical Specification Requirements in Appendix B of the FOL; and revise the Inter-Unit Fuel Transfer Technical Specifications in Appendix C. The proposed changes are consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The proposed changes would revise certain requirements contained within the IP3 FOL and TSs and remove the requirements that would no longer be applicable after IP3 is permanently shut down and defueled.

The existing IP3 Appendix A TSs contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing Appendix A TSs provide an appropriate level of control. However, the majority of the existing TSs are only applicable with the reactor in an operational mode. The LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the

Appendix A TSs are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents (DBAs) associated with a defueled facility.

The discussion below addresses each 10 CFR 50.92(c) no significant hazards consideration criterion and demonstrates that the proposed amendment does not constitute a significant hazard.

1. <u>Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   Response: No.

   The proposed amendment would not take effect until IP3 has permanently ceased operation, entered a permanently defueled condition, met the decay requirements established in the analysis of the Fuel Handling Accident (FHA), and implemented the NRC approved license amendment regarding administrative controls for the permanently defueled condition. The proposed amendment would modify the IP3 FOL and TSs in Appendices A through C by deleting the portions of the FOL and TSs that are no longer applicable to a permanently defueled facility, while modifying other portions to correspond to the permanently defueled condition. These proposed changes are consistent with the criteria set forth in 10 CFR 50.36 for the contents of TSs.

   Section 14 of the IP3 Updated Final Safety Analysis Report (UFSAR) describes the DBA and transient scenarios applicable to IP3 during power operations. After the reactor is in a permanently defueled condition, the spent fuel pit (SFP) and its cooling systems will be dedicated only to spent fuel storage. In this condition, the spectrum of credible accidents will be much smaller than for an operational plant. After the certifications are docketed for IP3 in accordance with 10 CFR 50.82(a)(1), and the consequent removal of authorization to operate the reactor or to place or retain fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2), the majority of the accident scenarios previously postulated in the UFSAR will no longer be possible and will be removed from the UFSAR under the provisions of 10 CFR 50.59.

   The deletion of TS definitions and rules of usage and application requirements that will not be applicable in a defueled condition has no impact on facility structures, systems, and components (SSCs) or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shut down and defueled status of IP3 has no impact on the remaining applicable DBAs.

   The removal of LCOs or SRs 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 do not affect the applicable DBAs previously evaluated since these DBAs are no longer applicable in the permanently defueled condition. The safety functions involving core reactivity control, reactor heat removal, reactor coolant system (RCS) inventory control, and containment integrity are no longer applicable at IP3 as a permanently shut down and defueled facility. The analyzed accidents involving damage to the RCS, main steam lines, reactor core, and the subsequent release of radioactive material will no longer be possible at IP3.

After IP3 permanently ceases operation, the future generation of fission products will cease and the remaining source term will decay. The radioactive decay of the irradiated fuel following shut down of the reactor will have reduced the consequences of the FHA below those previously analyzed.

The SFP water level, boron concentration, and fuel storage TSs are retained to preserve the current requirements for safe storage of irradiated fuel. The SFP cooling and make-up related equipment and support equipment (e.g., electrical power systems) are not required to be continuously available since there will be sufficient time to effect repairs, establish alternate sources of make-up flow, or establish alternate sources of cooling in the event of a loss of cooling and make-up flow to the SFP.

The deletion and modification of provisions of the administrative controls of the Appendix A TSs and the non-radiological environmental protection requirements in Appendix B do not directly affect the design of SSCs necessary for safe storage of irradiated fuel or the methods used for handling and storage of such fuel in the SFP. The changes do not affect any accidents applicable to the safe management of irradiated fuel or the permanently shut down and defueled condition of the reactor.

The probability of occurrence of previously evaluated accidents is not increased, since extended operation in a defueled condition will be the only condition allowed, and therefore bounded by the existing analyses.

Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible in a permanently defueled reactor. This significantly reduces the scope of applicable accidents.

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 IP3 FOL and Appendices A through C TSs have no impact on facility SSCs affecting the safe storage of irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of irradiated fuel itself. The removal of TSs 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 reactor will be permanently shut down and defueled and IP3 will no longer be authorized to operate the reactor.

The proposed deletion and modification of requirements of the IP3 FOL and Appendices A through C TSs do not affect systems credited in the accidents that remain applicable at IP3 in the permanently defueled condition. The proposed FOL and TSs will continue to require proper control and monitoring of safety significant parameters and activities.

The Appendix A TSs regarding SFP water level, boron concentration, and fuel storage are retained to preserve the current requirements for safe storage of irradiated fuel. The restriction on the SFP water level is fulfilled by normal operating conditions and preserves initial conditions assumed in the analyses of the postulated DBA.

The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding and spent fuel cooling). Since extended operation in a defueled condition will be the only condition allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

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.

Because the 10 CFR Part 50 license for IP3 will no longer authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel after the certifications required by 10 CFR 50.82(a)(1) are docketed for IP3 as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation are no longer credible. The only remaining credible accidents are the FHA and the accidental release of waste liquids or waste gas. The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses that impact the remaining DBAs.

The proposed amendment would modify the IP3 FOL and TSs in Appendices A through C by deleting the portions of the FOL and TSs that are no longer applicable to a permanently defueled facility, while modifying other portions to correspond to the permanently defueled condition. The requirements that are proposed to be deleted from the IP3 FOL and Appendix A TSs are not credited in the existing accident analyses for the remaining DBAs; and as such, do not contribute to the margin of safety associated with the accident analyses. Postulated DBAs involving the reactors will no longer be possible because the reactor will be permanently shut down and defueled and IP3 will no longer be authorized to operate the reactor.

The Appendix A TSs regarding SFP water level, boron concentration, and fuel storage are retained to preserve the current requirements for safe storage of irradiated fuel.

Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

3.3 PRECEDENT

The proposed changes to the IP3 FOL and Appendices A through C TSs are consistent with the intent of the license and accompanying PDTS issued to facilities that have been permanently shut down and defueled: (1) Fort Calhoun Station, for which an amendment was issued on March 6, 2018 (Reference 6); (2) Oyster Creek Nuclear Generating Station, for which an amendment was issued on October 26, 2018 (Reference 7); (3) San Onofre Nuclear Generating Station, Units 2 and 3, for which an amendment was issued on July 17, 2015 (Reference 8); (4) Crystal River Nuclear Plant, Unit 3, for which an amendment was issued on September 4, 2015 (Reference 9).

3.4 CONCLUSION

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4. ENVIRONMENTAL CONSIDERATIONS

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section 3.2 of this evaluation, the proposed amendment involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment does not involve any physical alterations to the facility configuration that could lead to a change in the type or amount of effluent release offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Entergy concludes that the proposed change meets the eligibility criteria for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5. REFERENCES

1. Entergy Nuclear Operations, Inc. (Entergy) letter to U.S. Nuclear Regulatory Commission (NRC), "Notification of Permanent Cessation of Power Operations," (Letter No. NL-17-021) (ADAMS Accession No. ML17044A004), dated February 8, 2017

2. NRC letter to Entergy, "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Issuance of Amendment Nos. 292 and 267 RE: Changes to Technical Specification Sections 1.1, 'Definitions'; 4.0, 'Design Features'; and 5.0, 'Administrative Controls,' for a Permanently Defueled Condition (EPID L-2019-LLA-0081)," (ADAMS Accession No. ML20071Q717), dated April 10, 2020

3. Entergy letter to NRC, "Proposed Technical Specifications (TS) Changes – Indian Point Nuclear Generating Unit 3 TS SR 3.7.7.2 and TS 3.7.6, Required Action A.1," (Letter No. NL-19-093) (ADAMS Accession No. ML19325E913), dated November 21, 2019

4. Holtec Decommissioning International, LLC (HDI) letter to NRC, "Post Shutdown Decommissioning Activities Report including Site-Specific Decommissioning Cost Estimate for Indian Point Nuclear Generating Units 1, 2, and 3," (ADAMS Accession No. ML19354A698), dated December 19, 2019

5. NRC letter to Entergy Services, LLC, "Arkansas Nuclear One, Units 1 and 2; Grand Gulf Nuclear Station, Unit 1; Indian Point Nuclear Generating Unit Nos. 2 and 3; Palisades Nuclear Plant; River Bend Station, Unit 1; Waterford Steam Electric Station, Unit 3 - Re: Issuance of Amendments to Adopt TSTF-529, 'Clarify Use And Application Rules' (EPID L- 2019-LLA-0013)," (ADAMS Accession No. ML19175A042), dated September 11, 2019

6. NRC letter to Omaha Public Power District, "Fort Calhoun Station, Unit 1 – Issuance of Amendment Re: Revised Technical Specifications to Align to Those Requirements for Decommissioning (CAC No. MF9567, EPID L-2017-LLA-0192)," (ADAMS Accession No. ML18010A087), dated March 6, 2018

7. NRC letter to Exelon Nuclear, "Oyster Creek Nuclear Generating Station – Issuance of Amendment Re: License Amendment Request for Proposed Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition (EPID L-2017-LLA-0395)," (ADAMS Accession No. ML18227A338), dated October 26, 2018

8. NRC letter to Southern California Edison Company, "San Onofre Nuclear Generating Station, Units 2 and 3 – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC Nos. MF3774 and MF3775)," (ADAMS Accession No. ML15139A390), dated July 17, 2015

9. NRC letter to Crystal River Nuclear Plant, "Crystal River Unit 3 Nuclear Generating Plant – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC No. MF3089)," (ADAMS Accession No. ML15224B286), dated September 4, 2019

Enclosure, Attachment 1

NL-20-033

Indian Point Nuclear Generating Station Unit 3
Mark-up of the Current Facility Operating License,
Appendices A through C Technical Specifications, and
Appendix A Technical Specifications Bases

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

ENTERGY NUCLEAR INDIAN POINT 3, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

RENEWED FACILITY ~~OPERATING~~ LICENSE

Renewed License No. DPR-64

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

   A. The application for a renewed license filed by Entergy Nuclear Indian Point 3, LLC (ENIP3) (the licensee) and Entergy Nuclear Operations, Inc. (ENO) (operator) for Indian Point Nuclear Generating Unit No. 3 (IP3 at the Indian Point Energy Center (IPEC) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will be maintained ~~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 renewed 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;

   D. ENIP3 and ENO are financially and technically qualified to engage in the activities authorized by this amendment; ~~Amdt. 203 11/27/00~~

   E. ENIP3 and ENO have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations; ~~Amdt. 203 11/27/00~~

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

   G. The receipt, possession and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70 including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31;

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

I. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); 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 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.

2. Accordingly, Renewed Facility ~~Operating~~ License No. DPR-64 is hereby issued to ENIP3 and ENO to read as follows:

maintained

A. This renewed license applies to the Indian Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by ENIP3 and ~~operated~~ by ENO. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "~~Final Facility Description and~~ Safety Analysis Report" as supplemented and amended, and the Environmental Report, as amended. ~~Amdt. 203 11/27/00~~

Defueled

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

and

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENIP3 to possess and use, and (b) ENO to possess, use ~~and operate,~~ the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license; ~~Amdt. 203 11/27/00~~

that was used

(2) ENO pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess, ~~and use,~~ at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation~~, as described in the ~~Final Facility Description and~~ Safety Analysis Report, as supplemented and amended; ~~Amdt. 203 11/27/00~~

Defueled

that were used

(3) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, at any time, any byproduct source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for ~~reactor~~ instrumentation and radiation monitoring equipment ~~calibration~~, and as fission detectors in amounts as required; ~~Amdt. 203 11/27/00~~

calibration of

that were used

are used in the calibration of

that were used

(4) ENO 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 without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus or components~~.~~ ; ~~Amdt. 203 11/27/00~~

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~ that were produced by the operation of the facility. ~~Amdt. 203 11/27/00~~

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

(1) ~~Maximum Power Level~~ Deleted per Amendment [###]

~~ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal (100% of rated power).~~

(2) Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. ~~268~~ ###, are hereby incorporated in the renewed license. ENO shall ~~operate~~ maintain the facility in accordance with the Technical Specifications.

~~(3) (DELETED)~~ ~~Amdt. 205 2-27-01~~

~~(4) (DELETED)~~ ~~Amdt. 205 2-27-01~~

D. (DELETED) ~~Amdt. 46 2-16-83~~

E. (DELETED) ~~Amdt. 37 5-14-81~~

F. This renewed license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

G. ENO 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 CFR 50.54(p). The combined set of plans[1] for the Indian Point Energy Center, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0," and was submitted by letter dated October 14, 2004, as supplemented by letter dated May 18, 2006.

ENO 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 ENO CSP was approved by License Amendment No. 243, as supplemented by changes approved by License Amendment Nos. 254, 260, and 263.

ENO has been granted Commission authorization to use "stand alone preemption authority" under Section 161A of the Atomic Energy Act, 42 U.S.C. 2201a with respect to the weapons described in Section II supplemented with Section III of Attachment 1 to its application submitted by letter dated August 20, 2013, as supplemented by letters dated November 21, 2013, and July 24, 2014, and citing letters dated April 27, 2011, and January 4, 2012. ENO shall fully implement and maintain in effect the provisions of the Commission-approved authorization.

Deleted per Amendment [###]

H. ~~ENO shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for Indian Point Nuclear Generating Unit No. 3 and as approved in NRC fire protection safety evaluations (SEs) dated September 21, 1973, March 6, 1979, May 2, 1980, November 18, 1982, December 30, 1982, February 2, 1984, April 16, 1984, January 7, 1987, September 9, 1988, October 21, 1991, April 20, 1994, January 5, 1995, and supplements thereto, subject to the following provision:~~

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

I. DELETED ~~Amdt. 205 2-27-01~~

J. DELETED ~~Amdt. 205 2-27-01~~

K. DELETED ~~Amdt. 49 5-25-84~~

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

L. DELETED ~~Amdt. 205 2-27-01~~

M. DELETED ~~Amdt. 205 2-27-01~~

Deleted per Amendment [###]

N. DELETED ~~Amdt. 49 5-25-84~~

O. ~~Evaluation, status and schedule for completion of balance of plant modifications as outlined in letter dated February 12, 1983, shall be forwarded to the NRC by January 1, 1984.~~ ~~Amdt. 47 5-27-83~~

P. ENIP3 and ENO shall take no action to cause Entergy Global Investments, Inc. or Entergy International Ltd. LLC, or their parent companies to void, cancel, or modify the $70 million contingency commitment to provide funding for the facility as represented in the application for approval of the transfer of the license from PASNY to ENIP3 and ENO, without the prior written consent of the Director, Office of Nuclear Reactor Regulation. ~~Amdt. 203 11/21/00~~

Q. DELETED

R. DELETED

S. DELETED

T. DELETED

U. DELETED

V. DELETED

W. For purposes of ensuring public health and safety, ENIP3, upon the transfer of this license to it, and upon transfer of decommissioning funds from PASNY to ENO, shall provide decommissioning funding assurance for the facility by the prepayment or equivalent method, to be held in a decommissioning trust fund for the facility, of no less than the amount required under NRC regulations at 10 CFR 50.75. Any amount held in any decommissioning trust maintained by ENO for the facility after the transfer of the facility license to ENIP3 may be credited towards the amount required under this paragraph.

X. ENIP3 shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for the transfer of this license to ENIP3 and ENO, as modified by the request to transfer decommissioning funds from PASNY, and the requirements of the order approving the transfer and order approving the transfer of decommissioning funds from PASNY to ENO, and consistent with the safety evaluations supporting such orders.

Deleted per Amendment [###]

AA. ~~The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications:~~ ~~Amdt. 205 2/27/01~~

~~1.~~ ~~This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee-controlled documents as described in Table R, "Relocated Technical Specifications from the CTS," and Table LA, "Removed Details and Less Restrictive Administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.~~

~~2.~~ ~~The following is a schedule for implementing surveillance requirements (SRs):~~

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

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

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

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

Deleted per Amendment [###]

AB. ~~With the reactor critical, Entergy shall maintain the reactor coolant system cold leg at a temperature (T_{cold}) greater than or equal to 525 °F. Entergy shall maintain a record of the cumulative time that the plant is operated with the reactor critical while T_{cold} is below 525 °F. Upon determination by Entergy that the cumulative time of plant operation with the reactor critical while T_{cold} is below 525 °F has exceeded one (1) year, Entergy must:~~

~~(a) within one (1) month, inform the NRC, in writing, and~~

~~(b) within six (6) months submit the results of an analysis of the impact of the operation with T_{cold} below 525 °F on the pressurized thermal shock reference temperature (RT_{pts}).~~

AC. Mitigation Strategy License Condition

The licensee shall 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

Deleted per Amendment [###]

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

AD. ~~Control Room Envelope Habitability~~

~~Upon implementation of Amendment No. 239 adopting TSTF-448, Revision 3 (as supplemented), the determination of control room envelope (CRE) unfiltered air inleakage as required by Technical Specification (TS) Surveillance Requirement (SR) 3. 7.11.4, in accordance with TS 5.5.16.c.(i), the assessment of CRE habitability as required by TS 5.5.16.c.(ii), and the measurement of CRE pressure as required by TS 5.5.16.d, shall be considered met. Following implementation:~~

(a) ~~The first performance of SR 3.7.11.4, in accordance with TS 5.5.16.c.(i), shall be within the specified Frequency of 6 years, plus the 18 month allowance of SR 3.0.2, as measured from February 1, 2005, the date of the most recent successful tracer gas test, as stated in the June 28, 2005, letter response to Generic Letter 2003-01.~~

(b) ~~The first performance of the periodic assessment of CRE habitability, TS 5.5.16.c.(ii), shall be within the next 9 months since the time period since the most recent successful tracer gas test is greater than 3 years.~~

(c) ~~The first performance of the periodic measurement of CRE pressure, TS 5.5.16.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from June 18, 2007, the date of the most recent successful pressure measurement test.~~

AE. ENO may transfer IP3 spent fuel to the IP2 spent fuel pit subject to the conditions listed in Appendix C. ENO is further authorized to transfer IP3 spent fuel into NRC approved storage casks for onsite storage by ENO and ENIP3.

AF. License Renewal License Conditions

(1) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d) and as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Units 2 and 3," (SER) and supplements to the SER, are collectively the "License Renewal UFSAR Supplement." The UFSAR Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in the UFSAR Supplement, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.

(2) The License Renewal UFSAR Supplement, as defined in license condition AF(1) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

a. The licensee shall implement those new programs and enhancements to existing programs no later than the date specified in the License Renewal UFSAR Supplement.

b. The licensee shall complete those activities no later than the date specified in the License Renewal UFSAR Supplement.

~~c. The licensee shall notify the NRC in writing within 30 days after having accomplished item (2)a above and include the status of those activities that have been or remain to be completed in item (2)b above.~~

3. This renewed license is effective as of the date of issuance, and ~~shall expire at midnight April 30, 2025~~.

until the Commission notifies the licensee in writing that the license is terminated

FOR THE NUCLEAR REGULATORY COMMISSION

PERMANENTLY DEFUELED

Ho K. Nieh, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Technical Specification Requirements
Appendix C – Inter-Unit Fuel Transfer Technical Specifications

Date of Issuance: ~~September 17, 2018~~

To Be Determined

PERMANENTLY DEFUELED

APPENDIX A

TO

FACILITY ~~OPERATING~~ LICENSE DPR-64

TECHNICAL SPECIFICATIONS AND BASES

FOR THE

INDIAN POINT 3 NUCLEAR GENERATING STATION UNIT NO. 3

WESTCHESTER COUNTY, NEW YORK

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

DOCKET NO. 50-286

Date of Issuance:
April 15, 1976

Amendment No. ~~203~~ ###

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SPENT FUEL PIT REQUIREMENTS

PERMANENTLY DEFUELED

(continued)

PERMANENTLY DEFUELED

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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.
~~ACTUATION LOGIC TEST~~	~~An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.~~
~~AXIAL FLUX DIFFERENCE (AFD)~~	~~AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.~~
CERTIFIED FUEL HANDLER (CFH)	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER Training and Retraining Program required by TS 5.3.2.
~~CHANNEL CALIBRATION~~	~~A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. 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 sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the~~

(continued)

~~1.1 Definitions~~

~~CHANNEL CALIBRATION (continued)~~

~~recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.~~

~~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 OPERATIONAL TEST (COT)~~

~~A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.~~

~~CORE ALTERATION~~

~~CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. 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 unit 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.~~

~~(continued)~~

~~1.1 Definitions (continued)~~

~~DOSE EQUIVALENT I-131~~

~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. If a specific isotope is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988.~~

~~DOSE EQUIVALENT XE-133~~

~~DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil".~~

~~L_a~~

~~The maximum allowable primary containment leakage rate, L_a, shall be 0.1% of primary containment air weight per day at the calculated peak containment pressure (P_a).~~

~~LEAKAGE~~

~~LEAKAGE shall be:~~

~~a. Identified LEAKAGE~~

~~1. LEAKAGE, such as that from pump seals or valve packing (except for leakage into closed systems and reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;~~

~~(Leakage into closed systems is leakage that can be accounted for and contained by a~~

~~(continued)~~

~~1.1 Definitions~~

~~LEAKAGE (continued)~~

~~system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and leakage past the safety injection pressure isolation valves are examples of reactor coolant system leakage into closed systems.)~~

~~2.~~ ~~LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or~~

~~3.~~ ~~Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);~~

~~b.~~ ~~Unidentified LEAKAGE~~

~~All LEAKAGE (except for leakage into closed systems and RCP seal water injection or leakoff) that is not identified LEAKAGE;~~

~~c.~~ ~~Pressure Boundary LEAKAGE~~

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

~~MASTER RELAY TEST~~ ~~A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.~~

~~MODE~~ ~~A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant loop temperature, and reactor~~

~~(continued)~~

1.0 USE AND APPLICATION

1.1 Definitions

~~MODE (continued)~~	~~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, but is not a CERTIFIED FUEL HANDLER.
~~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 FSAR Chapter 13, Initial Tests and Operations;~~ ~~b. Authorized under the provisions of 10 CFR 50.59; or~~ ~~c. Otherwise approved by the Nuclear Regulatory Commission.~~
~~QUADRANT POWER TILT RATIO (QPTR)~~	~~QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.~~
~~RATED THERMAL POWER (RTP)~~	~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3216 MWt.~~

(continued)

~~1.1 Definitions (continued)~~

~~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 rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and~~

~~b.~~ ~~In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power level.~~

~~SLAVE RELAY TEST~~

~~A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.~~

~~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~~ $\cancel{n}$ ~~Surveillance Frequency intervals, where~~ $\cancel{n}$ ~~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 reactor coolant.~~

~~(continued)~~

~~1.1 Definitions (continued)~~

~~TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)~~	~~A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.~~

~~Table 1.1-1 (page 1 of 1)~~
~~MODES~~

~~MODE~~	~~TITLE~~	~~REACTIVITY CONDITION (k_{eff})~~	~~% RATED THERMAL POWER[(a)]~~	~~AVERAGE REACTOR COOLANT TEMPERATURE (°F)~~
~~1~~	~~Power Operation~~	~~≥ 0.99~~	~~> 5~~	~~NA~~
~~2~~	~~Startup~~	~~≥ 0.99~~	~~≤ 5~~	~~NA~~
~~3~~	~~Hot Standby~~	~~< 0.99~~	~~NA~~	~~≥ 350~~
~~4~~	~~Hot Shutdown[(b)]~~	~~< 0.99~~	~~NA~~	~~350 > T_{avg} > 200~~
~~5~~	~~Cold Shutdown[(b)]~~	~~< 0.99~~	~~NA~~	~~≤ 200~~
~~6~~	~~Refueling[(c)]~~	~~NA~~	~~NA~~	~~NA~~

~~(a) Excluding decay heat.~~

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

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete ~~Conditions,~~ Required Actions~~, Completion Times,~~ and Surveillances~~, and Frequencies~~. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a ~~Condition, Completion Time,~~ Surveillance, ~~or Frequency,~~ only the first level of logic is used, and the logical connector is left justified with the statement of the ~~Condition, Completion Time,~~ Surveillance~~, or Frequency~~.

(continued)

1.2 Logical Connectors (continued)

EXAMPLE~~S~~

The following example~~s~~ illustrates 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.

(continued)

~~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.

handling and storage of spent nuclear fuel

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., ~~inoperable equipment or~~ variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility ~~unit~~ 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 features supported by an inoperable diesel generator, inoperable when the redundant required feature is inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied.~~

(continued)

1.3 Completion Times

DESCRIPTION (continued)

Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the ~~unit~~ facility is not within the LCO Applicability.

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

(continued)

~~1.3 Completion Times~~

~~DESCRIPTION (continued)~~

~~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 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.~~

~~(continued)~~

1.3 Completion Times (continued)

EXAMPLES

The following example~~s~~ Required Actions illustrate s the use of Completion Times with different types of ~~Conditions and changing Conditions~~.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A Spent fuel pit boron concentration not within limit. ~~Required Action and associated Completion Time not met.~~	~~B~~ A.1 Suspend movement of fuel assemblies in the spent fuel pit. ~~Be in MODE 3.~~ AND ~~B~~ A.2 Initiate action to restore spent fuel pit boron concentration to within limit. ~~Be in MODE 5.~~	~~6 hours~~ Immediately ~~36 hours~~ Immediately

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

The Required Actions of Condition ~~B~~ A are to immediately suspend movement of fuel assemblies in the spent fuel pit and initiate action to restore spent fuel pit boron concentration to within limit. ~~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.~~

(continued)

~~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.~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES~~

~~EXAMPLE 1.3-2 (continued)~~

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

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

~~(continued)~~

~~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.~~	~~72 hours~~ ~~72 hours~~

~~(continued)~~

~~1.3 Completion Times~~

~~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.~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-4~~

~~ACTIONS~~

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

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

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

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

~~(continued)~~

~~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.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Be in MODE 4.~~	~~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.~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES~~

~~EXAMPLE 1.3-5 (continued)~~

~~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.~~

~~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~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES~~

~~EXAMPLE 1.3-6 (continued)~~

~~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.~~

~~(continued)~~

~~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.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~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.~~

~~(continued)~~

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-7 (continued)~~

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

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~~. ~~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.~~
EXAMPLE~~S~~	The following exampl~~es 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~~. illustrates the type of Frequency statement that appears in the Technical Specifications (TS).

(continued)

1.4 Frequency

EXAMPLE~~S~~ (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify level is within limits. ~~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 facility ~~unit~~ is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility ~~unit~~ is in a ~~MODE or other~~ specified condition in the Applicability of the LCO, ~~and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3),~~ then SR 3.0.3 becomes applicable.

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

(continued)

~~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 25% 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.~~

~~(continued)~~

~~1.4 Frequency~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.4-3~~

~~SURVEILLANCE REQUIREMENTS~~

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

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

~~As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance 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 unit 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.~~

2.0 DELETED

Deleted

~~SLs~~
2.0

~~2.1 SLs~~

~~2.1.1 Reactor Core SLs~~

~~In MODES 1 and 2, the combination of THERMAL POWER, Reactor Vessel inlet temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:~~

~~2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-1 DNB correlations.~~

~~2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.~~

~~2.1.2 RCS Pressure SL~~

~~In MODES 1, 2, 3, 4, 5, and in MODE 6 when the reactor vessel head is on, the RCS pressure shall be maintained ≤ 2735 psig.~~

~~2.2 SL Violations~~

~~2.2.1 If SL 2.1.1 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 and LCO 3.0.8~~.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met~~, except as provided in LCO 3.0.5 and LCO 3.0.6~~.

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

~~LCO 3.0.3~~ ~~When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:~~

~~a.~~ ~~MODE 3 within 7 hours;~~

~~b.~~ ~~MODE 4 within 13 hours; and~~

~~c.~~ ~~MODE 5 within 37 hours.~~

~~Exceptions to this Specification are stated in the individual Specifications.~~

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

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

(continued)

~~3.0 LCO APPLICABILITY (continued)~~

~~LCO 3.0.4~~ ~~When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:~~

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

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

~~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 unit.~~

~~LCO 3.0.5~~ ~~Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.~~

~~(continued)~~

~~3.0 LCO APPLICABILITY (continued)~~

~~LCO 3.0.6~~

~~When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

~~When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~

~~LCO 3.0.7~~

~~Test Exception LCOs, such as 3.1.8, allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.~~

~~LCO 3.0.8~~

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

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

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

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

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the ~~MODES or other~~ specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on ~~inoperable equipment or~~ variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance ~~or as measured from the time a specified condition of the Frequency is met~~.

~~For Frequencies specified as "once," the above interval extension does not apply.~~

~~If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.~~

~~Exceptions to this Specification are stated in the individual Specifications.~~

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. ~~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.

(continued)

3.0 SR APPLICABILITY (continued)

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 unit.~~

SPENT FUEL PIT REQUIREMENTS

3.7 ~~PLANT SYSTEMS~~

3.7.14 Spent Fuel Pit Water Level

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

SPENT FUEL PIT REQUIREMENTS

3.7 ~~PLANT SYSTEMS~~

3.7.15 Spent Fuel Pit Boron Concentration

LCO 3.7.15 The Spent Fuel Pit boron concentration shall be ≥ 1000 ppm.

---------------------------NOTE-------------------------
During inter-unit transfer of fuel the spent fuel pit boron concentration must also meet Appendix C LCO 3.1.1, "Boron Concentration".

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

ACTIONS

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

SPENT FUEL PIT REQUIREMENTS

3.7 ~~PLANT SYSTEMS~~

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16

Fuel assemblies stored in the spent fuel pit shall be classified in accordance with Figure 3.7.16-1 based on initial enrichment and burnup; and,

Fuel assembly storage location within the spent fuel pit shall be restricted based on the Figure 3.7.16-1 classification as follows:

a. Fuel assemblies classified as Type 2 may be stored in any location in either Region 1 or Region 2;

b. Fuel assemblies classified as Type 1A, 1B or 1C shall be stored in Region 1;

c. Fuel assembly storage location within Region 1 shall be restricted as follows:

1. Type 1A assemblies may be stored anywhere in Region 1;

2. Type 1B assemblies may be stored anywhere in Region 1, except a Type 1B assembly shall not be stored face-adjacent to a Type 1C assembly;

3. Type 1C assemblies shall not be stored in Row 64 or in Column ZZ; and

4. Type 1C assemblies shall be stored in Region 1 locations where all face-adjacent locations are as follows:

a) occupied by Type 2 or Type 1A assemblies, or

b) occupied by non-fuel components, or

c) empty.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ~~NOTE~~ ~~LCO 3.0.3 is not applicable.~~ Initiate action to move fuel to restore compliance with LCO 3.7.16.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of each fuel assembly and that the storage location meets LCO 3.7.16 requirements.	Prior to storing the fuel assembly in the spent fuel pit

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

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

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

b. $k_{eff} \leq 0.95$ if assemblies are inserted in accordance with Technical Specification 3.7.16, Spent Fuel Assembly Storage~~.~~;

c. A nominal 9.075 inch center to center distance between fuel assemblies placed in the high density fuel storage racks (Region II);

d. A nominal 10.76 inch center to center distance between fuel assemblies placed in low density fuel storage racks (Region I)~~;~~.

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

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

~~b. $k_{eff} \leq 0.95$ under all possible moderation conditions (Credit may be taken for burnable integral neutron absorbers);~~

~~c. A nominal 20.5 inch center to center distance between fuel assemblies placed in the storage racks.~~

4.3.2 Drainage

The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below a nominal elevation of 88 ft.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear fuel.

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

b. The plant manager shall be responsible for overall safe maintenance of the facility and shall have control over those onsite activities necessary for safe storage and maintenance of nuclear fuel;

c. The corporate officer with direct responsibility for IP3 shall have corporate responsibility for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in maintaining and providing technical support to the facility to ensure safe management of nuclear fuel; and

d. The individuals who train the CERTIFIED FUEL HANDLERS, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 except as provided in the quality assurance program described or referenced in the ~~Updated FSAR.~~ DSAR;

b. Deleted ;

c. Quality assurance for effluent and environmental monitoring;

d. Deleted ~~Fire Protection Program implementation~~; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 ~~Primary Coolant Sources Outside Containment~~

~~Deleted~~

Deleted

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

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

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

  a. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

  b. For iodine-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit /facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each /facility ~~unit~~ to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

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

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Controls Program surveillance frequency.

Deleted

5.5.5 ~~Component Cyclic or Transient Limit~~

~~This program provides controls to track the FSAR, Section 4.1.5, cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

Deleted

5.5.6 ~~Reactor Coolant Pump Flywheel Inspection Program~~

~~This program shall provide for the inspection of each reactor coolant pump flywheel. The program shall include inspection frequencies and acceptance criteria. The inspection frequency will ensure that each reactor coolant pump flywheel is surface and volumetrically inspected at 20-year intervals.~~

(continued)

5.5 Programs and Manuals

Deleted

5.5.7 ~~Inservice Testing Program~~

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:~~

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

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

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

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

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

(continued)

5.5 Programs and Manuals

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:~~

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

~~2.~~ ~~Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the~~

(continued)

~~5.5.8 Steam Generator (SG) Program (continued)~~

~~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 per SG and 1 gpm through all SGs.~~

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

~~c. Provisions for SG tube plugging 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.~~

~~d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube to tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 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. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

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

~~2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new~~

~~(continued)~~

~~5.5.8~~ ~~Steam Generator (SG) Program (continued)~~

~~form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.~~

~~a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;~~

~~b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;~~

~~c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and~~

~~d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.~~

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

~~e. Provisions for monitoring operational primary to secondary LEAKAGE.~~

~~NOTE~~
~~Pages 5.0-16 through 5.0-19 are deleted.~~
~~Next page is 5.0-20.~~

~~(continued)~~

5.5 Programs and Manuals

5.5.9 ~~Secondary Water Chemistry Program~~

Deleted

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

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

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

~~c.~~ ~~Identification of process sampling points, which shall include monitoring the condenser hot wells 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 the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.~~

(continued)

5.5 Programs and Manuals

5.5.10 ~~Ventilation Filter Testing Program (VFTP)~~

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~~This program provides controls for implementation of required testing of the ventilation filter function for the Control Room Ventilation System and Containment Fan Cooler Units.~~

~~Applicable tests described in Specifications 5.5.10.a, 5.5.10.b, 5.5.10.c and 5.5.10.d shall be performed:~~

~~1) After 720 hours of charcoal adsorber use since the last test; and,~~

~~2) Every 24 months for the Control Room Ventilation System, and Containment Fan Cooler Units; and,~~

~~3) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; and,~~

~~4) After any structural maintenance on the system housing that could alter system integrity; and,~~

~~5) After significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.~~

~~SR 3.0.2 is applicable to the Ventilation Filter Testing Program.~~

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

~~a. Demonstrate for each system that an inplace test of the high efficiency particulate air (HEPA) filters shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.~~

~~Ventilation System~~	~~Removal Efficiency~~	~~Flowrate (cfm)~~	~~Reference Standard~~
~~Control Room Ventilation System~~	~~≥ 99%~~	~~80% to 120% of design accident rate~~	~~Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c~~
~~Containment Fan Cooler Units~~	~~≥ 99%~~	~~80% to 120% of design accident rate~~	~~Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.c~~

(continued)

~~5.5 Programs and Manuals~~

~~5.5.10 Ventilation Filter Testing Program (VFTP) (continued)~~

~~b. Demonstrate for each system that an inplace test of the charcoal adsorber shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the flowrate specified below.~~

~~Ventilation System~~	~~Removal Efficiency~~	~~Flowrate (cfm)~~	~~Reference Standard~~
~~Control Room Ventilation System~~	~~≥ 99%~~	~~80% to 120% of design accident rate~~	~~Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d~~
~~Containment Fan Cooler Units~~	~~≥ 99%~~	~~80% to 120% of design accident rate~~	~~Regulatory Guide 1.52, Rev 2, Sections C.5.a and C.5.d~~

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.10 Ventilation Filter Testing Program (VFTP) (continued)~~

~~c. Demonstrate for each system that a laboratory test of a sample of the charcoal adsorber shows the methyl iodide removal efficiency specified below when tested in accordance with ASTM D3803-1989, subject to clarification below, at a temperature of 86⁰F and a relative humidity of 95%.~~

~~Ventilation System~~	~~Methyl iodide removal efficiency (%):~~	~~ASTM D3803-1989 Clarification~~
~~Control Room Ventilation System~~	~~≥ 95.5~~	~~78 ft/min face velocity~~
~~Containment Fan Cooler Units~~	~~≥ 85~~	~~59 ft/min face velocity~~

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.10 Ventilation Filter Testing Program (VFTP) (continued)~~

~~d. Demonstrate for each system that the pressure drop across the combined HEPA filters, the demisters and prefilters (if installed), and the charcoal adsorbers is less than the value specified below when tested at the flowrate specified below.~~

~~Ventilation System~~	~~Delta P (inches wg)~~	~~Flowrate (cfm):~~
~~Control Room Ventilation System~~	~~6~~	~~≥ 90% of design accident rate~~
~~Containment Fan Cooler Units~~	~~6~~	~~≥ 90% of design accident rate~~

~~(continued)~~

5.5 Programs and Manuals

5.5.12 ~~Diesel Fuel Oil Testing Program~~

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

~~a.~~ ~~Verification of the acceptability of new fuel oil for use prior to addition to the DG fuel oil onsite storage tanks by determining that the fuel oil has:~~

~~1.~~ ~~Relative density within the limits of 0.83 to 0.89,~~

~~2.~~ ~~kinematic viscosity within the limits of 1.8 to 5.8, and~~

~~3.~~ ~~a clear and bright appearance with proper color~~

~~b1.~~ ~~Verification of the acceptability of the fuel oil in the onsite storage tanks and the reserve storage tanks every 92 days by verifying that the properties of the fuel oil in the tanks, other than those addressed in item a., are within limits for ASTM2D fuel oil. The sampling technique for the reserve storage tanks may deviate from ASTM D270-1975 in that only a bottom sample is required.~~

~~or~~

~~b2.~~ ~~Verification of the acceptability of each new fuel addition made subsequent to the last verification made in accordance with item b1. by verifying within 31 days following the addition that the properties of the new fuel oil, other than those properties addressed in item a. are within limits for ASTM 2D fuel oil.~~

~~c.~~ ~~Verification every 92 days that total particulate concentration of the fuel oil in the onsite and reserve storage tanks is less than or equal to 10 mg/l when tested in accordance with ASTM D-2276, Method A-2 or A-3. The sampling technique for the reserve storage tanks may deviate from ASTM D270-1975 in that only a bottom sample is required.~~

(continued)

5.5 Programs and Manuals

~~5.5.12~~ ~~Diesel Fuel Oil Testing Program (continued)~~

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

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the ~~updated FSAR~~ DSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that do not meet the criteria of Specification 5.5.13.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

~~5.5.14~~ ~~Safety Function Determination Program (SFDP)~~

~~This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may 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:~~

(continued)

~~5.5 Programs and Manuals~~

~~5.5.14~~ ~~Safety Function Determination Program (SFDP) (continued)~~

~~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, 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 the system(s) supported by the inoperable support system is also inoperable; or~~

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

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

~~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.~~

~~(continued)~~

~~5.5.15 Containment Leakage Rate Testing Program~~

~~A program shall be established to implement 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 2A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008, as modified by the following exceptions:~~

~~a. ANS 56.8-2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested.~~

~~b. The next Type A test to be performed after the March 2005 Type A test shall be performed no later than the plant restart after the Spring 2021 (3R21) RFO.~~

~~The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. P_a is the peak calculated containment internal pressure related to the design basis accident.~~

~~Leakage acceptance criteria are:~~

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

~~b. Air lock testing acceptance criteria are:~~

~~1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,~~

~~2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$,~~

~~c. Isolation Valve Seal Water System leakage rate acceptance criterion is $\leq$ 14,700 cc/hr at $\geq 1.1\ P_a$.~~

~~d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is $\leq$ 0.36 gpm per fan~~

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.15 Containment Leakage Rate Testing Program (continued)~~

~~cooler unit when pressurized at $\geq$ 1.1 P_a. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.~~

~~The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10CFR50, Appendix J.~~

~~The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.38 psig. The containment design pressure is 47 psig.~~

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

~~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 System (CRVS), 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 total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:~~

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.16 Control Room Envelope Habitability Program (continued)~~

~~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 CRVS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.~~

~~e. The quantitative limits on unfiltered air 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 analysis 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.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.

5.6.3 Radioactive Effluent Release Report

-----NOTE-----

A single submittal may be made for a multiple unit/facility station. The submittal shall combine sections common to all units/facilities at the station; however, for units/facilities with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit/facility.

The Radioactive Effluent Release Report covering the operation of the unit/facility 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 unit/facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR Part 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

~~5.6.4~~ ~~Not Used~~

~~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:~~

(continued)

~~5.6 Reporting Requirements~~

~~5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)~~

~~1. Specification 2.1, Safety Limits (SL);~~

~~2. Specification 3.1.1, Shutdown Margin;~~

~~3. Specification 3.1.3, Moderator Temperature Coefficient;~~

~~4. Specification 3.1.5, Shutdown Bank Insertion Limits;~~

~~5. Specification 3.1.6, Control Bank Insertion Limits;~~

~~6. Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));~~

~~7. Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor;~~

~~8. Specification 3.2.3, AXIAL FLUX DIFFERENCE (AFD);~~

~~9. Specification 3.3.1, Reactor Protection System Instrumentation;~~

~~10. Specification 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits; and~~

~~11. Specification 3.9.1, Boron Concentration.~~

~~b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:~~

~~1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Specifications 3.1.5, Shutdown Bank Insertion Limits, 3.1.6, Control Bank Insertion Limits, and 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor);~~

~~2a. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES, TOPICAL REPORT," September 1974 (W Proprietary). (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control);~~

~~2b. T. M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));~~

~~2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch~~

~~(continued)~~

~~5.6 Reporting Requirements~~

~~5.6.5~~ ~~CORE OPERATING LIMITS REPORT (COLR) (continued)~~

~~Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));~~

~~3.~~ ~~WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," March 1998 (Westinghouse Proprietary);~~

~~4.~~ ~~WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 (Specification 2.1, Safety Limits (SL)) and Specification 3.4.1, (RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits);~~

~~5.~~ ~~WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Specification 2.1, Safety Limits (SL));~~

~~6a.~~ ~~WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));~~

~~6b.~~ ~~WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code; Safety Injection into the Broken Loop and Cosi Condensation Model," July 1997 (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));~~

~~6c.~~ ~~WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));~~

~~7.~~ ~~WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor);~~

~~8.~~ ~~WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement", March 1997. (Specification 3.1.3, Moderator Temperature Coefficient;~~

~~9.~~ ~~WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON", August 2004. (Specification 3.1.1, Shutdown Margin, Specification 3.1.3, Moderator Temperature Coefficient, Specification 3.1.5, Shutdown Bank Insertion Limits, Specification 3.1.6, Control Bank Insertion Limits, Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)), Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor, Specification 3.2.3, Axial Flux Difference (AFD), Specification 3.9.1, Boron Concentration); and~~

~~(continued)~~

~~5.6 Reporting Requirements~~

~~5.6.5~~ ~~CORE OPERATING LIMITS REPORT (COLR) (continued)~~

~~10.~~ ~~WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986. (Specification 3.1.1, Shutdown Margin, Specification 3.1.3, Moderator Temperature Coefficient, Specification 3.1.5, Shutdown Bank Insertion Limits, Specification 3.1.6, Control Bank Insertion Limits, Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)), Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor, Specification 3.2.3, Axial Flux Difference (AFD), Specification 3.9.1, Boron Concentration).~~

~~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 (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~

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

~~5.6.6~~ ~~NOT USED~~

~~(continued)~~

~~5.6 Reporting Requirements~~

~~5.6.7~~ ~~Post Accident Monitoring Instrumentation (PAM) Report~~

~~When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the next 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.~~

~~5.6.8~~ ~~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.~~ ~~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 degradation mechanism,~~

~~f.~~ ~~The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG, and~~

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

APPENDIX B

TO

FACILITY ~~OPERATING~~ LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT 3 NUCLEAR

POWER PLANT

ENVIRONMENTAL TECHNICAL SPECIFICATION

REQUIREMENTS

PART I: NON-RADIOLOGICAL ENVIRONMENTAL PROTECTION PLAN

FACILITY LICENSE NO. DPR-64

DOCKET NUMBER 50-286

handling and storage of spent fuel
and maintenance

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:

facility is maintained

(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.

handling and storage of spent
fuel and maintenance of the

(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 SPDES permit.

3.0 Consistency Requirements

3.1 Plant Design and Operation

ENO may make changes in ~~station~~ facility design or operations 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 the ~~plant~~ facility 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, ENO shall prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity involves an unreviewed environmental question, ENO 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) or final supplemental environmental impact statement (FSEIS), as modified by the staff's testimony to the Atomic Safety and Licensing Boards, supplements to the FES or FSEIS, environmental impact appraisals, or in any decision of the Atomic Safety and Licensing Board;

* This provision does not relieve the ENO of the requirements of 10 CFR 50.59.

or (2) a significant change in effluents ~~or power level~~; 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.

ENO 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 provides a basis 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. ENO shall include as part of its Annual Environmental Protection Plan 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 4.1 of the Clean Water 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 ENIP3 and ENO 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. ENO 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

facility

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

the handling and storage of spent fuel and maintenance of the facility

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, unusual mortality or occurrence of any species protected by the Endangered Species Act of 1973, unusual fish kills, unusual 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

In accordance with Section 7(a) of the Endangered Species Act, the National Marine Fisheries Service (NMFS) issued a Biological Opinion related to the continued operation of IP2 and IP3 that pertains to shortnose sturgeon (*Acipenser brevirostrum*) and Atlantic sturgeon (*Acipenser oxyrinchus oxyrinchus*). The Biological Opinion includes an Incidental Take Statement with Reasonable and Prudent Measures that the NMFS has determined to be necessary or appropriate to minimize the amount or extent of incidental take and associated Terms and Conditions, which are non-discretionary and implement the Reasonable and Prudent Measures. The currently applicable Biological Opinion concludes that continued operation of IP2 and IP3 is not likely to jeopardize the continued existence of the listed species or to adversely affect the designated critical habitat of those species.

This Biological Opinion conservatively bounds the conditions that will occur in the permanently shut down and defueled condition.

5.0 Administrative Procedures

5.1 Review and Audit

ENO shall provide a 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 is utilized to achieve the independent review and audit function and results of the audits activities shall be maintained and made available for inspection.

and the handling and storage of spent fuel and maintenance of the facility

5.2 Records Retention

previous

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 the NRC on request.

facility

facility

Records of modifications to ~~plant~~ structures, systems and components determined to potentially affect the continued protection of the environment~~al~~ 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

Requests for changes in the Environmental Protection Plan shall include an assessment of the environmental impacts 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. This EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Protection Plan 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 non-radiological 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, ENO shall provide a detailed analysis of the data and a proposed course of action to alleviate the problem.

previous

and the handling and storage of spent fuel and maintenance of the facility

The Annual Environmental Protection Plan Report shall also include:

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

facility

(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.

5.4.2 Nonroutine Reports

facility conditions

A written report shall be submitted to the NRC within 30 days of occurrence of a 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 same time it is submitted to the other agency.

APPENDIX B

TO

FACILITY ~~OPERATING~~ LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT 3 NUCLEAR

POWER PLANT

ENVIRONMENTAL TECHNICAL SPECIFICATION

REQUIREMENTS

PART II: RADIOLOGICAL ENVIRONMENTAL

FACILITY LICENSE NO. DPR-64

DOCKET NUMBER 50-286

APPENDIX C

TO

FACILITY ~~OPERATING~~ LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART I: SPENT FUEL TRANSFER CANISTER AND TRANSFER CASK SYSTEM

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

SPENT FUEL SHIELDED TRANSFER CANISTER AND TRANSFER CASK SYSTEM

1.0 DESCRIPTION

The spent fuel transfer system consists of the following components: (1) a spent fuel shielded transfer canister (STC), which contains the fuel; (2) a transfer cask (HI-TRAC 100D) (hereafter referred to as HI-TRAC), which contains the STC during transfer operations; and (3) a bottom missile shield.

The STC and HI-TRAC are designed to transfer irradiated nuclear fuel assemblies from the Indian Point 3 (IP3) spent fuel pit to the Indian Point 2 (IP2) spent fuel pit. A fuel basket within the STC holds the fuel assemblies and provides criticality control. The shielded transfer canister provides the confinement boundary, water retention boundary, gamma radiation shielding, and heat rejection capability. The HI-TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The STC contains up to 12 fuel assemblies.

The STC is the confinement system for the fuel. It is a welded, multi-layer steel and lead cylinder with a welded base-plate and bolted lid. The inner shell of the canister forms an internal cylindrical cavity for housing the fuel basket. The outer surface of the canister inner shell is buttressed with lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are ¾ inch steel, 2 ¾ inch lead and ¾ inch steel, respectively. The canister closure incorporates two O-ring seals to ensure its confinement function. The confinement system consists of the canister inner shell, bottom plate, top flange, top lid, top lid O-ring seals, vent port seal and cover plate, and drain port seal and coverplate. The fuel basket, for the transfer of 12 Pressurized Water Reactor (PWR) fuel assemblies, is a fully welded, stainless steel, honeycomb structure with neutron absorber panels attached to the individual storage cell walls under stainless steel sheathing. The maximum gross weight of the fully loaded STC is 40 tons.

The HI-TRAC is a multi-layer steel and lead cylinder with a bolted bottom (or pool) and top lid. For the fuel transfer operation the HI-TRAC is fitted with a solid top lid, an STC centering assembly, and a bottom missile shield. The inner shell of the transfer cask forms an internal cylindrical cavity for housing the STC. The outer surface of the cask inner shell is buttressed with intermediate lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are ¾ inch steel, 2 ⅞ inch lead and 1 inch steel, respectively. An outside shell called the "water jacket" contains water for neutron shielding, with a minimum thickness of 5". The HI-TRAC bottom and top lids incorporate a gasket seal design to ensure its water confinement function. The water confinement system consists of the HI-TRAC inner shell, bottom lid, top lid, top lid seal, bottom lid seal, vent port seal, vent port cap and bottom drain plug.

The HI-TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The bottom missile shield is attached to the bottom of the HI-TRAC and provides tornado missile protection of the pool lid bolted joint. The HI-TRAC can withstand a tornado missile in other areas without the need for additional shielding. The STC centering assembly provides STC position control within the HI-TRAC and also acts as an internal impact limiter in the event of a non-mechanistic tipover accident.

2.0 CONDITIONS

2.1 OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, maintenance, and recovery from off normal conditions such as crane hang-up. The written operating procedures shall be consistent with the technical basis described in Chapter 10 of the Licensing Report (Holtec International Report HI-2094289).

2.2 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 8 of the Licensing Report (Holtec International Report HI-2094289).

2.3 PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A training exercise of the loading, closure, handling/transfer, and unloading, of the equipment shall be conducted prior to the first transfer. The training exercise shall not be conducted with irradiated fuel. The training exercise may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The training exercise shall include, but is not limited to the following:

a) Moving the STC into the IP3 spent fuel pool.
b) Preparation of the HI-TRAC for STC loading.
c) Selection and verification of specific fuel assemblies and non-fuel hardware to ensure type conformance.
d) Loading specific assemblies and placing assemblies into the STC (using a single dummy fuel assembly), including appropriate independent verification.
e) Remote installation of the STC lid and removal of the STC from the spent fuel pool.
f) Placement of the STC into the HI-TRAC with the STC centering assembly.
g) STC closure, establishment of STC water level with steam, verification of STC water level, STC leakage testing, and operational steps required prior to transfer, as applicable.
h) Establishment and verification of HI-TRAC water level.
i) Installation of the HI-TRAC top lid.
j) HI-TRAC closure, leakage testing, and operational steps required prior to transfer, as applicable.
k) Movement of the HI-TRAC with STC from the IP3 fuel handling building to the IP2 fuel handling building along the haul route with designated devices.
l) Moving the STC into the IP2 spent fuel pool.
m) Manual crane operations for bare STC movements including demonstration of recovery from a crane hang-up with the STC suspended from the crane.

APPENDIX C

TO

FACILITY ~~OPERATING~~ LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART II: TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

PERMANENTLY DEFUELED

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DELETED

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DELETED

B 2.0 ~~SAFETY LIMITS (SLs)~~

~~B 2.1.1 Reactor Core SLs~~

~~BASES~~

~~BACKGROUND~~

~~GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.~~

~~The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.~~

~~Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.~~

~~Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.~~

~~The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.~~

(continued)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.~~8~~ 2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the ~~unit~~ facility is in the ~~MODES or other~~ specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered~~, unless otherwise specified~~. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

~~There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.)~~

~~The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.~~

(continued)

BASES

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

~~The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."~~

~~The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The ACTIONS for not meeting a single LCO adequately manage any increase in plant risk, provided any unusual external conditions (e.g., severe weather, offsite power instability) are considered. In addition, the increased risk associated with simultaneous removal of multiple structures, systems, trains or components from service is assessed and managed in accordance with 10 CFR 50.65(a)(4). Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.~~

~~When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable.~~

~~In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.~~

(continued)

~~BASES~~

~~LCO 3.0.3~~

~~LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:~~

~~a.~~ ~~An associated Required Action and Completion Time is not met and no other Condition applies; or~~

~~b.~~ ~~The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.~~

~~This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. Planned entry into LCO 3.0.3 should be avoided. If it is not practicable to avoid planned entry into LCO 3.0.3, plant risk should be assessed and managed in accordance with 10 CFR 50.65(a)(4), and the planned entry into LCO 3.0.3 should have less effect on plant safety than other practicable alternatives.~~

~~Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to enter lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE.~~

(continued)

~~LCO 3.0.3 (continued)~~

~~This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.~~

~~A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:~~

~~a. The LCO is now met,~~

~~b. The LCO is no longer applicable,~~

~~c. A Condition exists for which the Required Actions have now been performed, or~~

~~d. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.~~

~~The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for entering the next lower MODE applies. If a lower MODE is entered in less time than allowed, however, the total allowable time to enter MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is entered in 2 hours, then the time allowed for entering MODE 4 is the next 11 hours, because the total time for entering MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to enter a lower MODE of operation in less than the total time allowed.~~

~~In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.~~

~~(continued)~~

~~BASES~~

~~LCO 3.0.3 (continued)~~

~~Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.14, "Spent Fuel Pit Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the Spent Fuel Pit." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in the Spent Fuel Pit" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.~~

~~LCO 3.0.4~~

~~LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with either LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.~~

~~LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered following entry into the MODE or other specified condition in the Applicability will permit continued operation within the MODE or other specified condition for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made and the Required Actions followed after entry into Applicability.~~

~~For example, LCO 3.0.4.a may be used when the Required Action to be entered states that an inoperable instrument channel must be placed in the trip condition within the Completion Time. Transition into a MODE or other specified condition in the Applicability may be made in accordance with LCO 3.0.4 and the channel is subsequently placed in the tripped condition within the Completion Time, which begins when the Applicability is entered. If the instrument channel cannot be placed in the tripped condition and the subsequent default~~

~~(continued)~~

~~LCO 3.0.4 (continued)~~

~~ACTION ("Required Action and associated Completion Time not met") allows the OPERABLE train to be placed in operation, use of LCO 3.0.4.a is acceptable because the subsequent ACTIONS to be entered following entry into the MODE include ACTIONS (place the OPERABLE train in operation) that permit safe plant operation for an unlimited period of time in the MODE or other specified condition to be entered.~~

~~LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met 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.~~

~~The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4(b), must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.~~

~~(continued)~~

~~LCO 3.0.4 (continued)~~

~~LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.~~

~~The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.~~

~~The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and the use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.~~

~~LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity), and may be applied to other Specifications based on NRC plant-specific approval.~~

~~(continued)~~

~~BASES~~

~~LCO 3.0.4 (continued)~~

~~The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.~~

~~The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.~~

~~Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.~~

~~Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.~~

~~(continued)~~

~~BASES~~

~~LCO 3.0.5~~

~~LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:~~

~~a. The OPERABILITY of the equipment being returned to service; or~~

~~b. The OPERABILITY of other equipment.~~

~~The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance. LCO 3.0.5 should not be used in lieu of other practicable alternatives that comply with Required Actions and that do not require changing the MODE or other specified conditions in the Applicability in order to demonstrate equipment is OPERABLE. LCO 3.0.5 is not intended to be used repeatedly.~~

~~An example of demonstrating equipment is OPERABLE with the Required Actions not met is opening a manual valve that was closed to comply with Required Actions to isolate a flowpath with excessive Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) leakage in order to perform testing to demonstrate that RCS PIV leakage is now within limit.~~

~~Examples of demonstrating equipment OPERABILITY include instances in which it is necessary to take an inoperable channel or trip system out of a tripped condition that was directed by a Required Action, if there is no Required Action Note for this purpose. An example of verifying OPERABILITY of equipment removed from service is taking a tripped channel out of the tripped condition to permit the logic to function and indicate the appropriate response during performance of required testing on the inoperable channel. Examples of demonstrating the OPERABILITY of other equipment are taking an inoperable channel or trip system out of the tripped condition 1) to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system, or 2)to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.~~

~~(continued)~~

~~LCO 3.0.5 (continued)~~

~~The administrative controls in LCO 3.0.5 apply in all cases to systems or components in Chapter 3 of the Technical Specifications, as long as the testing could not be conducted while complying with the Required Actions. This includes the realignment or repositioning of redundant or alternate equipment or trains previously manipulated to comply with ACTIONS, as well as equipment removed from service or declared inoperable to comply with ACTIONS.~~

~~LCO 3.0.6~~

~~LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.~~

~~When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.~~

~~However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry~~

~~(continued)~~

~~LCO 3.0.6 (continued)~~

~~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.~~

~~Specification 5.5.14, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6. Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required.~~

~~The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

~~(continued)~~

~~BASES~~

~~LCO 3.0.7~~

~~There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs, such as LCO 3.1.8, allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.~~

~~The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.~~

~~(continued)~~

~~BASES~~

~~LCO 3.0.8~~

~~LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.~~

~~If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.~~

~~LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.~~

~~LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.~~

~~LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing~~

~~(continued)~~

~~BASES~~

~~LCO 3.0.8 (continued)~~

~~Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.~~

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the ~~MODES or other~~ specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify ~~the OPERABILITY of systems and components, and~~ that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

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

~~a. The systems or components are known to be inoperable, although still meeting the SRs; or~~

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

facility

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

~~Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR.~~

~~This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.~~

(continued)

BASES

to restore variables within their specified limits.

variables that are outside their specified limits,

SR 3.0.1 (continued)

Surveillances~~, including Surveillances invoked by Required Actions,~~ do not have to be performed on ~~inoperable equipment~~ because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2~~, prior to returning equipment to OPERABLE sta~~tus.

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.~~

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances~~ and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval~~.

facility

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers ~~plant operating~~ conditions that may not be suitable for conducting the Surveillance (e.g., ~~transient conditions or~~ other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs.

~~The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations. Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."~~

(continued)

BASES

SR 3.0.2 (continued)

~~As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.~~

merely as a convenience

The provisions of SR 3.0.2 are not intended to be used repeatedly to extend Surveillance intervals ~~(other than those consistent with refueling intervals) or periodic Completion Time intervals~~ beyond those specified.

SR 3.0.3

completed

SR 3.0.3 establishes the flexibility to defer declaring ~~affected equipment inoperable or~~ an affected variable outside the specified limits when a Surveillance has not been ~~performed~~ within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

complete

This delay period provides adequate time to ~~perform~~ Surveillances that have been missed.

completion

This delay period permits the ~~performance~~ of a Surveillance before complying with Required Actions or other remedial measures that might preclude ~~performance~~ of the Surveillance.

completion

(continued)

BASES

SR 3.0.3 (continued)

The basis for this delay period includes consideration of facility ~~unit~~ conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. ~~When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.~~

~~SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.~~

~~SR 3.0.3 is only applicable if there is a reasonable expectation the associated equipment is OPERABLE or that variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or activities that might support the expectation that the Surveillance will be met when performed. An example of the use of SR 3.0.3 would be a relay contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed.~~

(continued)

SR 3.0.3 (continued)

~~For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.~~

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used ~~repeatedly~~ as a convenience to extend Surveillance intervals.

While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on ~~plant~~ facility risk (from delaying the Surveillance as well as any ~~plant~~ facility configuration changes required ~~or shutting the plant down~~ to perform the Surveillance) and impact on any analysis assumptions, in addition to ~~unit~~ facility conditions, planning, availability of personnel, and the time required to perform the Surveillance. ~~This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components~~

(continued)

BASES

SR 3.0.3 (continued)

~~should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action.~~ All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then ~~the equipment is considered inoperable or~~ the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then ~~the equipment is inoperable, or~~ the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a ~~MODE or other~~ specified condition in the Applicability.

variables within specified limits

This Specification ensures that ~~system and component OPERABILITY requirements and~~ variable limits are met before entry into ~~MODES or other~~ specified conditions in the Applicability for which these ~~systems and components ensure safe operation of the unit~~. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring ~~systems or components to OPERABLE status~~ before entering an associated ~~MODE or other~~ specified condition in the Applicability.

variables ensure safe handling and storage of spent fuel

~~A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to Surveillance not being met in accordance with LCO 3.0.4.~~

variables that are outside their specified limits

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a ~~MODE change or other~~ specified condition change. When a ~~system, subsystem, division, component, device, or~~ variable is ~~inoperable or~~ outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that Surveillances do not have to be performed on ~~inoperable equipment~~. When ~~equipment is inoperable~~, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing ~~MODES or~~

a variable is outside its specified limit

(continued)

BASES

SR 3.0.4 (continued)

~~other~~ specified conditions of the Applicability. ~~However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that my (or may not) apply to MODE or other specified condition changes.~~ SR 3.0.4 does not restrict changing ~~MODES or other~~ specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

~~The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.~~

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry in the ~~MODE or other~~ specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SPENT FUEL PIT REQUIREMENTS

B 3.7 ~~PLANT SYSTEMS~~

B 3.7.14 Spent Fuel Pit Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pit meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pit design and the Spent Fuel Cooling and Cleanup System is given in the FSAR, Section 9.5 (Ref. 1). The assumptions of the fuel handling accident are given in the FSAR, Section 14.2 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel pit meets the assumptions of the fuel handling accident described in FSAR, Section 14.2 (Ref. 2). The resultant 2 hour thyroid dose per person at the exclusion area boundary satisfies the 10 CFR 50.67 (Ref. 3) limits.

According to Reference 2, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks.

The Spent Fuel Pit water level satisfies Criteria 2 and 3 of 10 CFR 50.36.

LCO

The spent fuel pit water level is required to be 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel storage and movement within the spent fuel pit.

(continued)

BASES

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pit, since the potential for a release of fission products exists.

ACTIONS

A.1

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pit water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pit is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

~~If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient spent fuel pit water is available in the event of a fuel handling accident. The water level in the spent fuel pit must be checked periodically. The 7 day Frequency is appropriate because the volume in the spent fuel pit is normally stable. Water level changes are controlled by ~~plant~~ procedures and are acceptable based on operating experience.

~~During refueling operations, the level in the spent fuel pit is normally in equilibrium with the refueling canal and reactor cavity, and the level in the refueling reactor cavity is checked daily in accordance with SR 3.9.6.1.~~

(continued)

B 3.7 ~~PLANT SYSTEMS~~

B 3.7.15 Spent Fuel Pit Boron Concentration

BASES

BACKGROUND

In the Maximum Density Rack (MDR) design, the spent fuel storage pool is divided into two separate and distinct regions. The layout of the IP3 MDR is shown in Figure B 3.7.16-1. As shown in Figure B 3.7.16-1, Region 1 (Columns SS-ZZ, Rows 35-64) includes 240 storage positions and Region 2 (Columns A-RR, Rows 1-34) includes 1105 storage positions. Region 1 is analyzed for storage of high-enrichment and low-burnup fuel. Region 2 is analyzed for storage of fuel with either higher burnup or lower enrichment. Each region has been separately analyzed for close packed storage when all cells in that region contain fuel of the highest reactivity stored in accordance with LCO 3.7.16, Spent Fuel Assembly Storage. This analysis is the basis for the restrictions on fuel storage locations established by LCO 3.7.16.

Limits, based on a combination of initial enrichment and burnup, are used to determine if a fuel assembly must be stored in region 1 or if the fuel assembly may be stored in either region 1 or region 2. Fuel with the highest initial enrichments are subject to additional restrictions even when stored in region 1. Fuel assemblies with an initial enrichment > 5.0 wt% U-235 cannot be stored in the spent fuel pit in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel pit normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded when fuel storage locations, enrichment and burnup are in conformance with analysis assumptions as specified in LCO 3.7.16. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, because only a single accident need be considered at one time. For example, the accident

(continued)

BASES

LCO (continued)

confirms that there are no misloaded fuel assemblies. With no misloaded fuel assemblies and unborated water, the spent fuel pit design is sufficient to maintain the core at $K_{eff} \leq 0.95$.

The LCO is modified by a Note that states that during inter-unit transfer of fuel the spent fuel pit boron concentration must also meet Appendix C LCO 3.1.1, "Boron Concentration". This requirement ensures that the criticality analysis of the fuel within the Shielded Transfer Canister remains bounding.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pit, until a complete spent fuel pit verification has been performed following the last movement of fuel assemblies in the spent fuel pit. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1 and A.2.2

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the concentration of boron in the spent fuel pit is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the Spent Fuel Pit fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

(continued)

BASES

~~ACTIONS~~

~~A.1, A.2.1 and A.2.2 (continued)~~

~~If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pit is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 31 day Frequency is appropriate because no major replenishment of spent fuel pit water is expected to take place over such a short period of time. This SR is not required to be met or performed if a spent fuel pit verification for conformance with LCO 3.7.16, Figures 3.7.16-1 and B 3.7.16-1, has been performed on all fuel assemblies since the last verification following the last movement of fuel assemblies in the spent fuel pit.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SER related to Amendment 173 to Facility Operating License No. DPR-64, Indian Point Nuclear Generating Unit No. 3, April 15, 1997.

3. Criticality Analysis of the Indian Point 3 Fresh and Spent Fuel Racks, Westinghouse Commercial Nuclear Fuel Division, October, 1996.

SPENT FUEL PIT REQUIREMENTS

B 3.7 ~~PLANT SYSTEMS~~

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

In the Maximum Density Rack (MDR) design, the spent fuel pit (SFP) is divided into two separate and distinct regions. The layout of the IP3 MDR is shown in Figure B 3.7.16-1, IP3 Maximum Density Spent Fuel Pit Racks, Regions and Indexing. As shown in Figure B 3.7.16-1, Region 1 (i.e., Columns SS-ZZ, Rows 35-64) includes 240 storage positions and Region 2 (i.e., Columns A-RR, Rows 1-34) includes 1105 storage positions. Region 1 is analyzed for storage of high-enrichment and low-burnup fuel. Region 2 is analyzed for storage of fuel with either higher burnup or lower enrichment. Each region has been separately analyzed for close packed storage when all cells in that region contain fuel of the highest reactivity that is allowed by this LCO. This analysis is the basis for the restrictions on fuel storage locations established by this LCO.

Prior to storage in the spent fuel pit, fuel assemblies are classified as to the level of reactivity based on the initial enrichment and burnup. This classification is made using Figure 3.7.16-1, "Fuel Assembly Classification for Storage in the Spent Fuel Pit". This classification is used to determine in which region a particular fuel assembly may be stored and if additional restrictions must be applied to the assemblies in adjacent locations. Figure 3.7.16-1, "Fuel Assembly Classification for Storage in the Spent Fuel Pit", is used to classify each assembly into one of the following categories based on initial U-235 enrichment and burnup:

Type 2 assemblies are the least reactive assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the domain labeled Type 2 in Figure 3.7.16-1. Type 2 assemblies may be stored in any location in Region 1 or Region 2 of Figure B 3.7.16-1.

Type 1A assemblies are more reactive than Type 2 assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the domain labeled

(continued)

BASES

LCO

Fuel assemblies stored in the spent fuel pit are classified in accordance with Figure 3.7.16-1 based on initial enrichment and burnup which is indicative of fuel assembly reactivity. Based on this classification, fuel assembly storage location within the spent fuel pit and storage location relative to other assemblies is restricted in accordance with the rules established by this LCO.

Fuel assemblies with an initial enrichment > 5.0 wt% U-235 are not shown on Figure 3.7.16-1 because fuel assemblies with this enrichment cannot be stored in the spent fuel pit in accordance with limits established in Technical Specification Section 4.3.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pit.

ACTIONS

A.1

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the configuration of fuel assemblies stored in the spent fuel pit is not in accordance with this LCO, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with this LCO.

~~If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~

(continued)

Enclosure, Attachment 2

NL-20-033

Indian Point Nuclear Generating Station Unit 3
Re-typed (Clean) Facility License,
Appendix A Permanently Defueled Technical Specifications,
Appendices B and C Technical Specifications, and
Appendix A Permanently Defueled Technical Specifications Bases

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

ENTERGY NUCLEAR INDIAN POINT 3, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

RENEWED FACILITY LICENSE

Renewed License No. DPR-64

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

   A. The application for a renewed license filed by Entergy Nuclear Indian Point 3, LLC (ENIP3) (the licensee) and Entergy Nuclear Operations, Inc. (ENO) (operator) for Indian Point Nuclear Generating Unit No. 3 (IP3 at the Indian Point Energy Center (IPEC) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will be maintained 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 renewed 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;

   D. ENIP3 and ENO are financially and technically qualified to engage in the activities authorized by this amendment;

   E. ENIP3 and ENO have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations;

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

   G. The receipt, possession and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70 including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31;

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

I. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); 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 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.

2. Accordingly, Renewed Facility License No. DPR-64 is hereby issued to ENIP3 and ENO to read as follows:

A. This renewed license applies to the Indian Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by ENIP3 and maintained by ENO. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "Defueled Safety Analysis Report" as supplemented and amended, and the Environmental Report, as amended.

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

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENIP3 to possess and use, and (b) ENO to possess and use the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;

(2) ENO pursuant to the Act and 10 CFR Part 70, to possess, at any time, special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended;

(3) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, at any time, any byproduct source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment, and that were used as fission detectors in amounts as required;

(4) ENO 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 without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus or components;

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.

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

(1) <u>Deleted per Amendment [###]</u>

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. ###, are hereby incorporated in the renewed license. ENO shall maintain the facility in accordance with the Technical Specifications.

D. (DELETED)

E. (DELETED)

F. This renewed license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

G. ENO 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 CFR 50.54(p). The combined set of plans[1] for the Indian Point Energy Center, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0," and was submitted by letter dated October 14, 2004, as supplemented by letter dated May 18, 2006.

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1 The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

ENO 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 ENO CSP was approved by License Amendment No. 243, as supplemented by changes approved by License Amendment Nos. 254, 260, and 263.

ENO has been granted Commission authorization to use "stand alone preemption authority" under Section 161A of the Atomic Energy Act, 42 U.S.C. 2201a with respect to the weapons described in Section II supplemented with Section III of Attachment 1 to its application submitted by letter dated August 20, 2013, as supplemented by letters dated November 21, 2013, and July 24, 2014, and citing letters dated April 27, 2011, and January 4, 2012. ENO shall fully implement and maintain in effect the provisions of the Commission-approved authorization.

H. Deleted per Amendment [###]

I. <u>DELETED</u>

J. <u>DELETED</u>

K. <u>DELETED</u>

L. <u>DELETED</u>

M. <u>DELETED</u>

N. <u>DELETED</u>

O. Deleted per Amendment [###]

P. ENIP3 and ENO shall take no action to cause Entergy Global Investments, Inc. or Entergy International Ltd. LLC, or their parent companies to void, cancel, or modify the $70 million contingency commitment to provide funding for the facility as represented in the application for approval of the transfer of the license from PASNY to ENIP3 and ENO, without the prior written consent of the Director, Office of Nuclear Reactor Regulation.

Q. <u>DELETED</u>

R. <u>DELETED</u>

S. <u>DELETED</u>

T. <u>DELETED</u>

U. <u>DELETED</u>

V. <u>DELETED</u>

W. For purposes of ensuring public health and safety, ENIP3, upon the transfer of this license to it, and upon transfer of decommissioning funds from PASNY to ENO, shall provide decommissioning funding assurance for the facility by the prepayment or equivalent method, to be held in a decommissioning trust fund for the facility, of no less than the amount required under NRC regulations at 10 CFR 50.75. Any amount held in any decommissioning trust maintained by ENO for the facility after the transfer of the facility license to ENIP3 may be credited towards the amount required under this paragraph.

X. ENIP3 shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for the transfer of this license to ENIP3 and ENO, as modified by the request to transfer decommissioning funds from PASNY, and the requirements of the order approving the transfer and order approving the transfer of decommissioning funds from PASNY to ENO, and consistent with the safety evaluations supporting such orders.

AA. Deleted per Amendment [###]

AB. Deleted per Amendment [###]

AC. Mitigation Strategy License Condition

The licensee shall 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

AD. Deleted per Amendment [###]

AE. ENO may transfer IP3 spent fuel to the IP2 spent fuel pit subject to the conditions listed in Appendix C. ENO is further authorized to transfer IP3 spent fuel into NRC approved storage casks for onsite storage by ENO and ENIP3.

AF. License Renewal License Conditions

(1) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d) and as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Units 2 and 3," (SER) and supplements to the SER, are collectively the "License Renewal UFSAR Supplement." The UFSAR Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in the UFSAR Supplement, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.

(2) The License Renewal UFSAR Supplement, as defined in license condition AF(1) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

a. The licensee shall implement those new programs and enhancements to existing programs no later than the date specified in the License Renewal UFSAR Supplement.

b. The licensee shall complete those activities no later than the date specified in the License Renewal UFSAR Supplement.

3. This renewed license is effective as of the date of issuance, and until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

Ho K. Nieh, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A - Permanently Defueled Technical Specifications
Appendix B - Environmental Technical Specification Requirements
Appendix C - Inter-Unit Fuel Transfer Technical Specifications

Date of Issuance: To Be Determined

APPENDIX A

TO

FACILITY LICENSE DPR-64

PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS AND BASES

FOR THE

INDIAN POINT 3 NUCLEAR GENERATING STATION UNIT NO. 3

WESTCHESTER COUNTY, NEW YORK

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

DOCKET NO. 50-286

Date of Issuance:
April 15, 1976

Amendment No. ###

TABLE OF CONTENTS

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.
CERTIFIED FUEL HANDLER (CFH)	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER training and retraining program required by TS 5.3.2.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.

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 Required Actions and Surveillances. 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 indentations of the logical connectors.

When logical connectors are used to state a Surveillance, only the first level of logic is used, and the logical connector is left justified with the statement of the Surveillance.

EXAMPLE

The following example illustrates 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.

1.0 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 handling and storage of spent nuclear fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.
EXAMPLE	The following example illustrates the use of Completion Times with different Required Actions.

1.3 Completion Time

EXAMPLE (continued)

EXAMPLE 1.3-1

ACTIONS

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

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

The Required Actions of Condition A are to immediately suspend movement of fuel assemblies in the spent fuel pit and initiate action to restore spent fuel pit boron concentration within limit.

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.

EXAMPLE

The following example illustrates the type of Frequency statement that appears in the Technical Specifications (TS).

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify level is within limits.	12 hours

Example 1.4-1 contains the type of SR 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 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 a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a 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 specified condition or the LCO is considered not met (in accordance with SR 3.0.1).

2.0 DELETED

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. 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.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the 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 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.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4

Entry into a 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.

3.7 SPENT FUEL PIT REQUIREMENTS

3.7.14 Spent Fuel Pit Water Level

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pit water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in the spent fuel pit.	Immediately

SURVEILLANCE REQUIREMENTS

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

3.7 SPENT FUEL PIT REQUIREMENTS

3.7.15 Spent Fuel Pit Boron Concentration

LCO 3.7.15 The spent fuel pit boron concentration shall be ≥ 1000 ppm.

---NOTE---
During inter-unit transfer of fuel the spent fuel pit boron concentration must also meet Appendix C LCO 3.1.1, "Boron Concentration."

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pit boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pit.	Immediately
	AND	
	A.2.1 Initiate action to restore spent fuel pit boron concentration to within limit.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel pit verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pit boron concentration is within limit.	31 days

3.7 SPENT FUEL PIT REQUIREMENTS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 Fuel assemblies stored in the spent fuel pit shall be classified in accordance with Figure 3.7.16-1 based on initial enrichment and burnup; and,

Fuel assembly storage location within the spent fuel pit shall be restricted based on the Figure 3.7.16-1 classification as follows:

a. Fuel assemblies classified as Type 2 may be stored in any location in either Region 1 or Region 2;

b. Fuel assemblies classified as Type 1A, 1B or 1C shall be stored in Region 1;

c. Fuel assembly storage location within Region 1 shall be restricted as follows:

1. Type 1A assemblies may be stored anywhere in Region 1;
2. Type 1B assemblies may be stored anywhere in Region 1, except a Type 1B assembly shall not be stored face-adjacent to a Type 1C assembly;
3. Type 1C assemblies shall not be stored in Row 64 or in Column ZZ; and
4. Type 1C assemblies shall be stored in Region 1 locations where all face-adjacent locations are as follows:

   a) occupied by Type 2 or Type 1A assemblies, or

   b) occupied by non-fuel components, or

   c) empty.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move fuel to restore compliance with LCO 3.7.16.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of each fuel assembly and that the storage location meets LCO 3.7.16 requirements.	Prior to storing the fuel assembly in the spent fuel pit

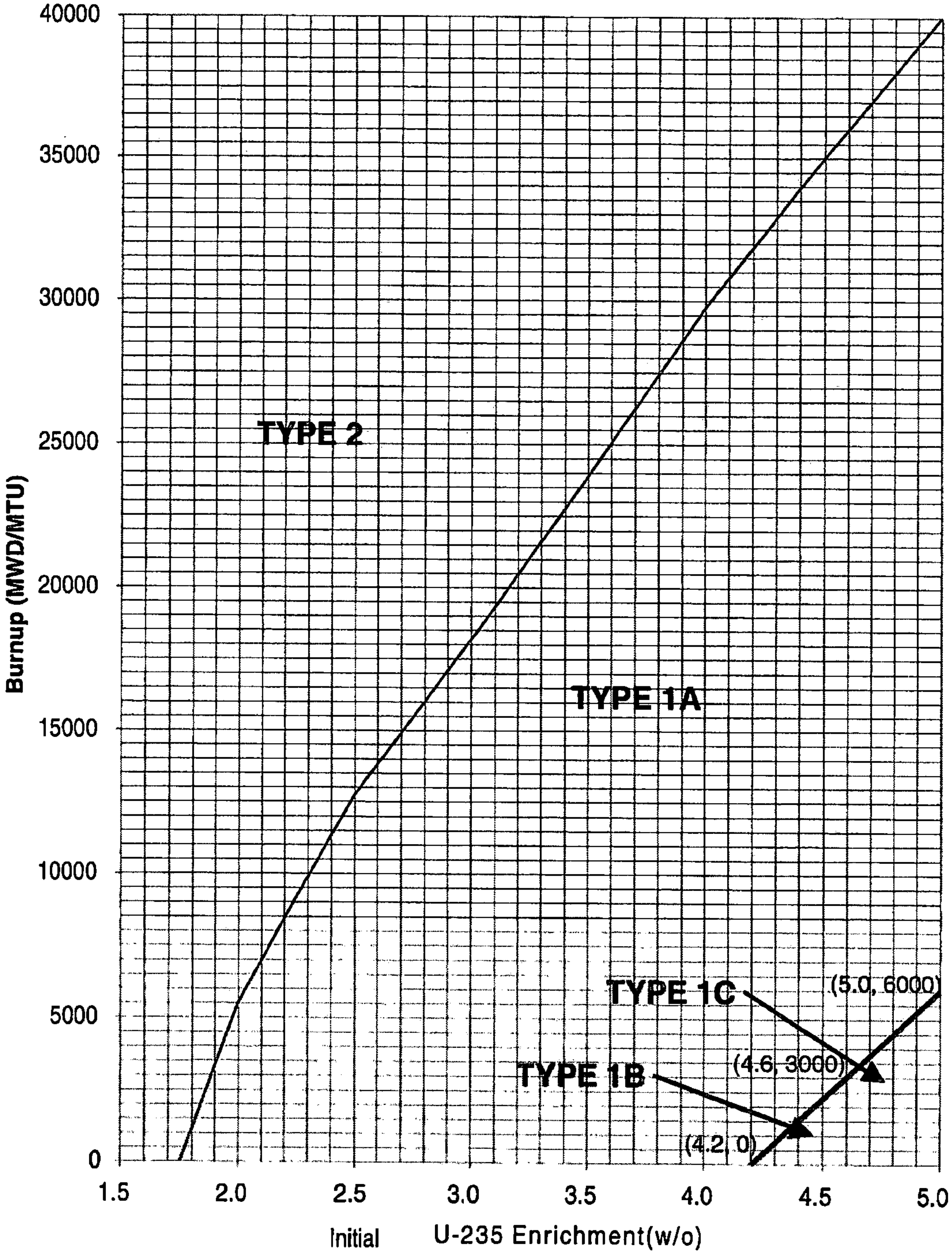

Figure 3.7.16-1 (Page 1 of 1)

Fuel Assembly Classification for Storage in the Spent Fuel Pit

4.0 DESIGN FEATURES

4.1 Site Location

Indian Point 3 is located on the east bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York. The site is approximately 24 miles north of the New York City boundary line. The nearest city is Peekskill which is 2.5 miles northeast of Indian Point.

The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone as defined in 10 CFR 100.3 is 350 meters and 1100 meters, respectively.

4.2 Deleted

4.3 Fuel Storage

4.3.1 Criticality

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

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

b. keff ≤ 0.95 if assemblies are inserted in accordance with Technical Specification 3.7.16, Spent Fuel Assembly Storage;

c. A nominal 9.075 inch center to center distance between fuel assemblies placed in the high density fuel storage racks (Region II);

d. A nominal 10.76 inch center to center distance between fuel assemblies placed in low density fuel storage racks (Region I);

4.3.2 Drainage

The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below a nominal elevation of 88 ft.

4.3.3 Capacity

The spent fuel pit is designed and shall be maintained with a storage capacity limited to no more than 1345 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

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

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

5.1.2 The shift manager (SM) shall be responsible for the shift command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear fuel.

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

b. The plant manager shall be responsible for overall safe maintenance of the facility and shall have control over those onsite activities necessary for safe storage and maintenance of nuclear fuel;

c. The corporate officer with direct responsibility for IP3 shall have corporate responsibility for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in maintaining and providing technical support to the facility to ensure safe management of nuclear fuel; and

d. The individuals who train the CERTIFIED FUEL HANDLERS, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2 Organization

5.2.2 Facility Staff

The facility staff organization shall include the following:

a. Each duty shift shall be composed of at least one shift manager and one NON-CERTIFIED OPERATOR. The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.

At least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room when nuclear fuel is stored in the spent fuel pool.

b. Shift crew composition may be less than the minimum requirement of 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of the following conditions are met:

1) No fuel movements are in progress;

2) No movement of loads over fuel are in progress; and

3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.

c. An individual qualified in radiation protection procedures shall be on site during fuel handling operations and during movement of heavy loads over the fuel storage racks. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Not Used.

e. The shift manager shall be a CERTIFIED FUEL HANDLER.

f. Deleted.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the IPEC Quality Assurance Program Manual (QAPM).

5.3.2 An NRC approved training and retraining program for CERTIFIED FUEL HANDLERS shall be maintained.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 except as provided in the quality assurance program described or referenced in the DSAR;

b. Deleted;

c. Quality assurance for effluent and environmental monitoring;

d. Deleted; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

c. Licensee initiated changes to the ODCM:

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

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

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

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

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Deleted

5.5 Programs and Manuals

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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

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

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

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

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

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

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. 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 limited to the following:

   a. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b. For iodine-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit/facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit/facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

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

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Controls Program surveillance frequency.

5.5.5 through 5.5.10 Deleted

5.5 Programs and Manuals

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

The program shall include:

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

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank shall be limited to less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

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

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.12 Deleted

5.5 Programs and Manuals

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the DSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that do not meet the criteria of Specification 5.5.13.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Not Used

5.6.2 Annual Radiological Environmental Operating Report

---NOTE---
A single submittal may be made for a multiple unit/facility station. The submittal should combine sections common to all units/facilities at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit/facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.

5.6.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit/facility station. The submittal shall combine sections common to all units/facilities at the station; however, for units/facilities with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit/facility.

The Radioactive Effluent Release Report covering the operation of the unit/facility 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 unit/facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR Part 50.36a and 10 CFR 50, Appendix I, Section IV.B.I.

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 facility 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; 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 supervisor, 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 facility 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 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; 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.

APPENDIX B TO

FACILITY LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3) AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT 3 NUCLEAR

POWER PLANT

ENVIRONMENTAL TECHNICAL SPECIFICATION

REQUIREMENTS

PART I: NON-RADIOLOGICAL ENVIRONMENTAL PROTECTION PLAN

FACILITY LICENSE NO. DPR-64

DOCKET NUMBER 50-286

INDIAN POINT NUCLEAR GENERATING PLANT
UNIT 3

ENVIRONMENTAL TECHNICAL SPECIFICATION REQUIREMENTS
PART I: NON-RADIOLOGICAL ENVIRONMENTAL PROTECTION PLAN

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during handling and storage of spent fuel and maintenance of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is maintained 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 handling and storage of spent fuel and maintenance of the facility 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 SPDES permit.

3.0 Consistency Requirements

3.1 Plant Design and Operation

ENO may make changes in facility design or operations 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 the facility 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, ENO shall prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity involves an unreviewed environmental question, ENO 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) or final supplemental environmental impact statement (FSEIS), as modified by the staff's testimony to the Atomic Safety and Licensing Boards, supplements to the FES or FSEIS, environmental impact appraisals, or in any decision of the Atomic Safety and Licensing Board;

* This provision does not relieve the ENO of the requirements of 10 CFR 50.59.

or (2) a significant change in effluents; 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.

ENO 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 provides a basis 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. ENO shall include as part of its Annual Environmental Protection Plan 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 4.1 of the Clean Water 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 ENIP3 and ENO 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. ENO 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 facility 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 the handling and storage of spent fuel and maintenance of the facility 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, unusual mortality or occurrence of any species protected by the Endangered Species Act of 1973, unusual fish kills, unusual 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

In accordance with Section 7(a) of the Endangered Species Act, the National Marine Fisheries Service (NMFS) issued a Biological Opinion related to the continued operation of IP2 and IP3 that pertains to shortnose sturgeon (*Acipenser brevirostrum*) and Atlantic sturgeon (*Acipenser oxyrinchus oxyrinchus*). The Biological Opinion includes an Incidental Take Statement with Reasonable and Prudent Measures that the NMFS has determined to be necessary or appropriate to minimize the amount or extent of incidental take and associated Terms and Conditions, which are non-discretionary and implement the Reasonable and Prudent Measures. The currently applicable Biological Opinion concludes that continued operation of IP2 and IP3 is not likely to jeopardize the continued existence of the listed species or to adversely affect the designated critical habitat of those species. This Biological Opinion conservatively bounds the conditions that will occur in the permanently shut down and defueled condition.

Entergy shall adhere to the requirements within the Incidental Take Statement of the currently applicable Biological Opinion. Changes to the Biological Opinion, including the Incidental Take Statement, Reasonable and Prudent Measures, and Terms and Conditions contained therein, must be preceded by consultation between the NRC, as the authorizing agency, and the NMFS.

5.0 Administrative Procedures

5.1 Review and Audit

ENO shall provide a 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 is utilized to achieve the independent review and audit function and results of the audits activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of previous operation and the handling and storage of spent fuel and maintenance of the facility shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to the NRC on request.

Records of modifications to facility structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the facility. 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

Requests for changes in the Environmental Protection Plan shall include an assessment of the environmental impacts 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. This EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Protection Plan 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 non-radiological environmental monitoring reports, and an assessment of the observed impacts of the previous plant operation and the handling and storage of spent fuel and maintenance of the facility on the environment. If harmful effects or evidence of trends towards irreversible damage to the environment are observed, ENO shall provide a detailed analysis of the data and a proposed course of action to alleviate the problem.

The Annual Environmental Protection Plan 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 facility 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.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and facility conditions, (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 same time it is submitted to the other agency.

APPENDIX B

TO

FACILITY LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT 3 NUCLEAR

POWER PLANT

ENVIRONMENTAL TECHNICAL SPECIFICATION

REQUIREMENTS

PART II: RADIOLOGICAL ENVIRONMENTAL

FACILITY LICENSE NO. DPR-64

DOCKET NUMBER 50-286

APPENDIX C

TO

FACILITY LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART I: SPENT FUEL TRANSFER CANISTER AND TRANSFER CASK SYSTEM

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

SPENT FUEL SHIELDED TRANSFER CANISTER AND TRANSFER CASK SYSTEM

1.0 DESCRIPTION

The spent fuel transfer system consists of the following components: (1) a spent fuel shielded transfer canister (STC), which contains the fuel; (2) a transfer cask (HI-TRAC 100D) (hereafter referred to as HI-TRAC), which contains the STC during transfer operations; and (3) a bottom missile shield.

The STC and HI-TRAC are designed to transfer irradiated nuclear fuel assemblies from the Indian Point 3 (IP3) spent fuel pit to the Indian Point 2 (IP2) spent fuel pit. A fuel basket within the STC holds the fuel assemblies and provides criticality control. The shielded transfer canister provides the confinement boundary, water retention boundary, gamma radiation shielding, and heat rejection capability. The HI-TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The STC contains up to 12 fuel assemblies.

The STC is the confinement system for the fuel. It is a welded, multi-layer steel and lead cylinder with a welded base-plate and bolted lid. The inner shell of the canister forms an internal cylindrical cavity for housing the fuel basket. The outer surface of the canister inner shell is buttressed with lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are ¾ inch steel, 2 ¾ inch lead and ¾ inch steel, respectively. The canister closure incorporates two O-ring seals to ensure its confinement function. The confinement system consists of the canister inner shell, bottom plate, top flange, top lid, top lid O-ring seals, vent port seal and cover plate, and drain port seal and coverplate. The fuel basket, for the transfer of 12 Pressurized Water Reactor (PWR) fuel assemblies, is a fully welded, stainless steel, honeycomb structure with neutron absorber panels attached to the individual storage cell walls under stainless steel sheathing. The maximum gross weight of the fully loaded STC is 40 tons.

The HI-TRAC is a multi-layer steel and lead cylinder with a bolted bottom (or pool) and top lid. For the fuel transfer operation the HI-TRAC is fitted with a solid top lid, an STC centering assembly, and a bottom missile shield. The inner shell of the transfer cask forms an internal cylindrical cavity for housing the STC. The outer surface of the cask inner shell is buttressed with intermediate lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are ¾ inch steel, 2 ½ inch lead and 1 inch steel, respectively. An outside shell called the "water jacket" contains water for neutron shielding, with a minimum thickness of 5". The HI-TRAC bottom and top lids incorporate a gasket seal design to ensure its water confinement function. The water confinement system consists of the HI-TRAC inner shell, bottom lid, top lid, top lid seal, bottom lid seal, vent port seal, vent port cap and bottom drain plug.

The HI-TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The bottom missile shield is attached to the bottom of the HI-TRAC and provides tornado missile protection of the pool lid bolted joint. The HI-TRAC can withstand a tornado missile in other areas without the need for additional shielding. The STC centering assembly provides STC position control within the HI-TRAC and also acts as an internal impact limiter in the event of a non-mechanistic tipover accident.

2.0 CONDITIONS

2.1 OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, maintenance, and recovery from off normal conditions such as crane hang-up. The written operating procedures shall be consistent with the technical basis described in Chapter 10 of the Licensing Report (Holtec International Report HI-2094289).

2.2 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 8 of the Licensing Report (Holtec International Report HI-2094289).

2.3 PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A training exercise of the loading, closure, handling/transfer, and unloading, of the equipment shall be conducted prior to the first transfer. The training exercise shall not be conducted with irradiated fuel. The training exercise may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The training exercise shall include, but is not limited to the following:

a) Moving the STC into the IP3 spent fuel pool.
b) Preparation of the HI-TRAC for STC loading.
c) Selection and verification of specific fuel assemblies and non-fuel hardware to ensure type conformance.
d) Loading specific assemblies and placing assemblies into the STC (using a single dummy fuel assembly), including appropriate independent verification.
e) Remote installation of the STC lid and removal of the STC from the spent fuel pool.
f) Placement of the STC into the HI-TRAC with the STC centering assembly.
g) STC closure, establishment of STC water level with steam, verification of STC water level, STC leakage testing, and operational steps required prior to transfer, as applicable.
h) Establishment and verification of HI-TRAC water level.
i) Installation of the HI-TRAC top lid.
j) HI-TRAC closure, leakage testing, and operational steps required prior to transfer, as applicable.
k) Movement of the HI-TRAC with STC from the IP3 fuel handling building to the IP2 fuel handling building along the haul route with designated devices.
l) Moving the STC into the IP2 spent fuel pool.
m) Manual crane operations for bare STC movements including demonstration of recovery from a crane hang-up with the STC suspended from the crane.

APPENDIX C

TO

FACILITY LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART II: TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

Amendment No. ###

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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
SR 3.0.1	SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable. Surveillances do not have to be performed on variables that are outside their specified limits, because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2 to restore variables within their specified limits.
SR 3.0.2	SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances. SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., other ongoing Surveillance or maintenance activities). The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a convenience to extend Surveillance intervals beyond those specified.

BASES

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed.

This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals.

While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on facility risk (from delaying the Surveillance as well as any facility configuration changes required to perform the Surveillance) and impact on any analysis assumptions, in addition to facility conditions, planning, availability of personnel, and the time required to perform the Surveillance. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

BASES

SR 3.0.3 (continued)

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that variable limits are met before entry into specified conditions in the Applicability for which these variables ensure safe handling and storage of spent fuel. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring variables within specified limits before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a specified condition change. When a variable is outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that Surveillances do not have to be performed on variables that are outside their specified limits. When a variable is outside its specified limit, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. SR 3.0.4 does not restrict changing specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry in the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.7 SPENT FUEL PIT REQUIREMENTS

B 3.7.14 Spent Fuel Pit Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel pit meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the spent fuel pit design and the Spent Fuel Cooling and Cleanup System is given in the FSAR, Section 9.5 (Ref. 1). The assumptions of the fuel handling accident are given in the FSAR, Section 14.2 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The minimum water level in the spent fuel pit meets the assumptions of the fuel handling accident described in FSAR, Section 14.2 (Ref. 2). The resultant 2 hour thyroid dose per person at the exclusion area boundary satisfies the 10 CFR 50.67 (Ref. 3) limits. According to Reference 2, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. The spent fuel pit water level satisfies Criteria 2 and 3 of 10 CFR 50.36.
LCO	The spent fuel pit water level is required to be 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel storage and movement within the spent fuel pit.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel pit, since the potential for a release of fission products exists.

BASES

ACTIONS

A.1

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pit water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pit is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient spent fuel pit water is available in the event of a fuel handling accident. The water level in the spent fuel pit must be checked periodically. The 7 day Frequency is appropriate because the volume in the spent fuel pit is normally stable. Water level changes are controlled by procedures and are acceptable based on operating experience.

REFERENCES

1. FSAR, Section 9.5.
2. FSAR, Section 14.2.
3. 10 CFR 50.67.
4. Safety Evaluation Report (SER) for IP3 Amendment.

B 3.7 SPENT FUEL PIT REQUIREMENTS

B 3.7.15 Spent Fuel Pit Boron Concentration

BASES

BACKGROUND

In the Maximum Density Rack (MDR) design, the spent fuel storage pool is divided into two separate and distinct regions. The layout of the IP3 MDR is shown in Figure B 3.7.16-1. As shown in Figure B 3.7.16-1, Region 1 (Columns SS-ZZ, Rows 35-64) includes 240 storage positions and Region 2 (Columns A-RR, Rows 1-34) includes 1105 storage positions. Region 1 is analyzed for storage of high-enrichment and low-burnup fuel. Region 2 is analyzed for storage of fuel with either higher burnup or lower enrichment. Each region has been separately analyzed for close packed storage when all cells in that region contain fuel of the highest reactivity stored in accordance with LCO 3.7.16, Spent Fuel Assembly Storage. This analysis is the basis for the restrictions on fuel storage locations established by LCO 3.7.16.

Limits, based on a combination of initial enrichment and burnup, are used to determine if a fuel assembly must be stored in Region 1 or if the fuel assembly may be stored in either Region 1 or Region 2. Fuel with the highest initial enrichments are subject to additional restrictions even when stored in Region 1. Fuel assemblies with an initial enrichment > 5.0 wt% U-235 cannot be stored in the spent fuel pit in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel pit normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded when fuel storage locations, enrichment and burnup are in conformance with analysis assumptions as specified in LCO 3.7.16. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, because only a single accident need be considered at one time. For example, the accident

BASES

BACKGROUND (continued)

scenarios include movement of fuel from Region 1 to Region 2, or accidental misloading of a fuel assembly in Region 1. This event could increase the potential for criticality of the spent fuel pit. To mitigate these postulated criticality related accidents, boron concentration is verified by SR 3.7.15.1 to be within the limits specified in this LCO prior to movement of fuel assemblies in the spent fuel pit. Safe operation of the MDR with no movement of assemblies is achieved by controlling the location of each assembly in accordance with LCO 3.7.16, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.15.1.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of either of the two regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in both regions. The postulated accidents are basically of two types. A fuel assembly could be incorrectly transferred from Region 1 to Region 2 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded storage rack. This could have a small positive reactivity effect in the region. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is described in References 2 and 3.

The concentration of dissolved boron in the spent fuel pit satisfies Criterion 2 of 10 CFR 50.36.

LCO

The spent fuel pit boron concentration is required to be ≥ 1000 ppm. The specified concentration of dissolved boron in the spent fuel pit preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pit until a spent fuel pit verification confirms that there are no misloaded fuel assemblies. With no misloaded fuel assemblies and unborated water, the spent fuel pit design is sufficient to maintain the core at $K_{eff} \leq 0.95$.

BASES

LCO (continued)

The LCO is modified by a Note that states that during inter-unit transfer of fuel the spent fuel pit boron concentration must also meet Appendix C LCO 3.1.1, "Boron Concentration." This requirement ensures that the criticality analysis of the fuel within the Shielded Transfer Canister remains bounding.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pit, until a complete spent fuel pit verification has been performed following the last movement of fuel assemblies in the spent fuel pit. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1 and A.2.2

When the concentration of boron in the spent fuel pit is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel pit fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pit is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 31 day Frequency is appropriate because no major replenishment of spent fuel pit water is expected to take place over such a short period of time. This SR is not required to be met or performed if a spent fuel pit verification for conformance with LCO 3.7.16, Figures 3.7.16-1 and B 3.7.16-1, has been performed on all fuel assemblies since the last verification following the last movement of fuel assemblies in the spent fuel pit.

BASES

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SER related to Amendment 173 to Facility Operating License No. DPR-64, Indian Point Nuclear Generating Unit No. 3, April 15, 1997.

3. Criticality Analysis of the Indian Point 3 Fresh and Spent Fuel Racks, Westinghouse Commercial Nuclear Fuel Division, October, 1996.

B 3.7 SPENT FUEL PIT REQUIREMENTS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

In the Maximum Density Rack (MDR) design, the spent fuel pit (SFP) is divided into two separate and distinct regions. The layout of the IP3 MDR is shown in Figure B 3.7.16-1, IP3 Maximum Density Spent Fuel Pit Racks, Regions and Indexing. As shown in Figure B 3.7.16- 1, Region 1 (i.e., Columns SS-ZZ, Rows 35-64) includes 240 storage positions and Region 2 (i.e., Columns A-RR, Rows 1-34) includes 1105 storage positions. Region 1 is analyzed for storage of high-enrichment and low-burnup fuel. Region 2 is analyzed for storage of fuel with either higher burnup or lower enrichment. Each region has been separately analyzed for close packed storage when all cells in that region contain fuel of the highest reactivity that is allowed by this LCO. This analysis is the basis for the restrictions on fuel storage locations established by this LCO.

Prior to storage in the spent fuel pit, fuel assemblies are classified as to the level of reactivity based on the initial enrichment and burnup. This classification is made using Figure 3.7.16-1, "Fuel Assembly Classification for Storage in the Spent Fuel Pit." This classification is used to determine in which region a particular fuel assembly may be stored and if additional restrictions must be applied to the assemblies in adjacent locations. Figure 3.7.16-1, "Fuel Assembly Classification for Storage in the Spent Fuel Pit," is used to classify each assembly into one of the following categories based on initial U-235 enrichment and burnup:

<u>Type 2</u> assemblies are the least reactive assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the domain labeled Type 2 in Figure 3.7.16-1. Type 2 assemblies may be stored in any location in Region 1 or Region 2 of Figure B 3.7.16-1.

<u>Type 1A</u> assemblies are more reactive than Type 2 assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the domain labeled

BASES

BACKGROUND (continued)

Type 1A in Figure 3.7.16-1. Type 1A assemblies must be stored in Region 1 of Figure B 3.7.16-1 but may be stored in any location in Region 1.

Type 1B assemblies are more reactive than Type 1A assemblies and include any assembly with an initial enrichment > 4.2 but ≤ 4.6 wt% U-235 with a burnup that places the assembly in the domain labeled Type 1B in Figure 3.7.16-1. Type 1B assemblies must be stored in Region 1 of Figure B 3.7.16-1 but may be stored in any location in Region 1 except in locations that are face-adjacent to a Type 1C assembly.

Type 1C assemblies are the most reactive bundles permitted in accordance with Specification 4.3, Fuel Storage. Type 1C assemblies include any assembly with an initial enrichment > 4.6 but ≤ 5.0 wt% U-235 with a burnup that places the assembly in the domain labeled Type 1C on Figure 3.7.16-1. Type 1C assemblies must be stored in Region 1 of Figure B 3.7.16-1. Type 1C assemblies cannot be stored in Row 64 or in Column ZZ. Additionally, Type 1C assemblies must be stored in a location where all face-adjacent locations are as follows:

a) occupied by Type 2 or Type 1A assemblies;

b) occupied non-fuel components; or,

c) empty.

Fuel assemblies with an initial enrichment > 5.0 wt% U-235 are not shown on Figure 3.7.16-1 and cannot be stored in the spent fuel pit in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel pit normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded and fuel storage locations, enrichment and burnup are in conformance with analysis assumptions and this LCO. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions because only a single accident need be considered at one time. For example, the accident scenarios include movement of a Type 1C fuel assembly from Region 1 to Region 2, or accidental misloading of a fuel assembly in Region 1. These events

BASES

BACKGROUND (continued)

could increase the potential for criticality in the spent fuel pit. To mitigate these postulated criticality related accidents, boron concentration is verified to be within the limits specified in LCO 3.7.15, Spent Fuel Pit Boron Concentration, prior to movement of any fuel assembly. Safe operation of the SFP with no movement of assemblies is achieved by controlling the location of each assembly in accordance with the accompanying LCO. However, prior to movement of an assembly, it is necessary to perform SR 3.7.15.1 (i.e., verification that the spent fuel pit boron concentration is within limit).

APPLICABLE SAFETY ANALYSES

The restrictions on the placement of fuel assemblies within the spent fuel pit are based on initial enrichment and burnup which is indicative of fuel assembly reactivity. Storage locations are then restricted to ensure the k_{eff} of the spent fuel pit will always remain < 0.95, assuming the pool to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure 3.7.16-1 may not be stored in accordance with Specification 4.3.1.1 in Section 4.3.

The hypothetical accidents can only take place during or as a result of the movement of an assembly (References 2 and 3). For these accident occurrences, the presence of soluble boron in the spent fuel storage pit (controlled by LCO 3.7.15, "Spent Fuel Pit Boron Concentration") prevents criticality in both regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pit satisfies Criterion 2 of 10 CFR 50.36.

LCO

Fuel assemblies stored in the spent fuel pit are classified in accordance with Figure 3.7.16-1 based on initial enrichment and burnup which is indicative of fuel assembly reactivity. Based on this classification, fuel assembly storage location within the spent fuel pit and storage location relative to other assemblies is restricted in accordance with the rules established by this LCO.

Fuel assemblies with an initial enrichment > 5.0 wt% U-235 are not shown on Figure 3.7.16-1 because fuel assemblies with this enrichment cannot be stored in the spent fuel pit in accordance with limits established in Technical Specification Section 4.3.

BASES

APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel pit.
ACTIONS	A.1 When the configuration of fuel assemblies stored in the spent fuel pit is not in accordance with this LCO, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with this LCO.
SURVEILLANCE REQUIREMENTS	SR 3.7.16.1 This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly in each location is in accordance with the accompanying LCO.
REFERENCES	1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A). 2. SER related to Amendment 173 to Facility Operating License No. DPR-64, Indian Point Nuclear Generating Unit No. 3, April 15, 1997. 3. Criticality Analysis of the Indian Point 3 Fresh and Spent Fuel Racks, Westinghouse Commercial Nuclear Fuel Division, October, 1996.

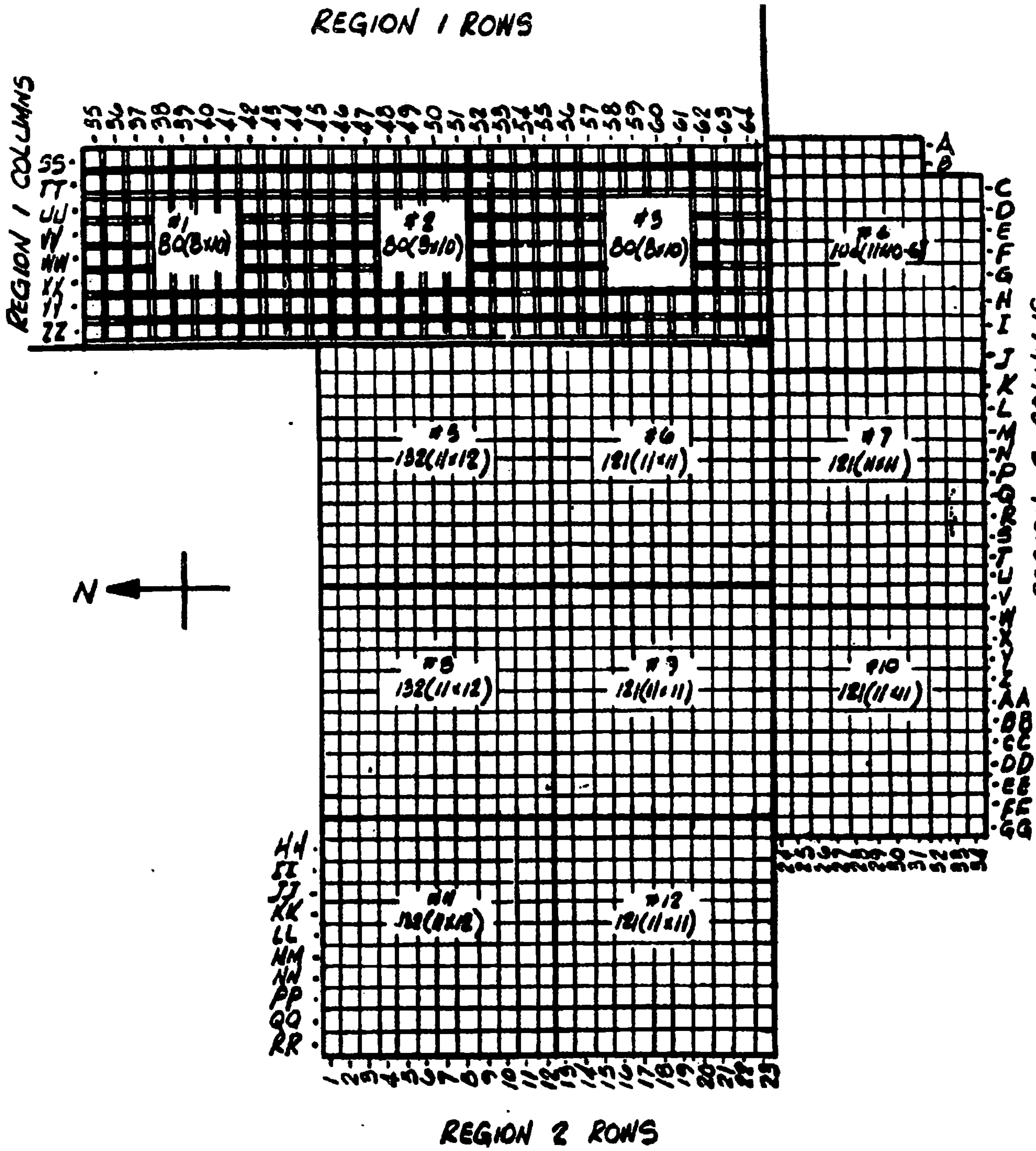

Figure B 3.7.16-1 (Page 1 of 1)
Maximum Density Spent Fuel Pit (SFP)
Racks, Regions and Indexing