

**Douglas B. Huey**  
Director – Site Performance Improvement

419-321-8408  
Fax: 419-321-7582

February 5, 2019  
L-18-242

10 CFR 50.90

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

**SUBJECT:**

Davis-Besse Nuclear Power Station, Unit No. 1  
Docket No. 50-346, License No. NPF-3  
License Amendment Request – Proposed License and Associated Technical Specification  
Changes for Permanently Defueled Condition

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the Renewed Facility Operating License No. NPF-3 for Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The proposed changes would revise the license and associated Technical Specifications consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

By letter dated April 25, 2018 (Accession No. ML18115A007), FENOC certified to the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.82(a)(1)(i) and 10 CFR 50.4(b)(8) that power operation will cease at DBNPS by May 31, 2020.

Once the certifications of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel is docketed for DBNPS, in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the license will no longer authorize reactor operation or emplacement or retention of fuel into the reactor vessel. In support of the permanently shutdown and defueled condition, a revision to the DBNPS license and associated technical specifications is proposed in accordance with 10 CFR 50.51(b) and 10 CFR 50.36(c)(6).

The basis for this proposed license amendment request is that certain license conditions and associated technical specifications may be revised or removed to reflect the permanently defueled condition. In general, the changes propose the elimination of items applicable in operating conditions where fuel is placed in the reactor vessel. The enclosure to this letter provides a detailed description and evaluation of the proposed changes, including markups of the current pages.



FENOC requests review and approval of this proposed amendment by January 31, 2020 to support the current schedule for transition to a permanently defueled facility. The amendment shall be implemented within 30 days following FENOC submittal of the certification required by 10 CFR 50.82(a)(1)(ii) that the DBNPS fuel has been permanently removed from the reactor vessel.

FENOC has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92, "Issuance of amendment."

The NRC is currently reviewing a supporting licensing action to change the organization, staffing, and training requirements contained in Section 5.0, "Administrative Controls," of the DBNPS Technical Specifications that was submitted for approval by letter dated October 22, 2018 (Accession No. ML18295A289).

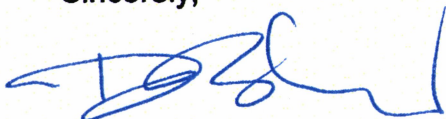
The NRC is also currently reviewing the licensing actions listed below that are unrelated to the proposed changes for reasons described in the enclosure:

- A proposed change to the fire protection program in License Condition 2.C(4) and the removal of associated Technical Specification 5.4.1.d was submitted for approval by letter dated December 16, 2015 (Accession No. ML15350A314) and supplemented by letters dated March 7, 2016, July 28, 2016, December 16, 2016, January 17, 2017, June 16, 2017, October 9, 2017, April 2, 2018, September 11, 2018, and November 20, 2018 (Accession Nos. ML16067A195, ML16210A422, ML16351A330, ML17017A504, ML17170A000, ML17284A190, ML18094A798, ML18254A073, and ML18324A677 respectively).
- A proposed change to the support agreement in License Condition 3.B was submitted for approval by letter dated May 18, 2017 (Accession No. ML17138A381) and supplemented by letter dated August 23, 2018 (Accession No. ML18235A194).

There are no regulatory commitments contained in this submittal. If there are any questions, or if additional information is required, please contact Mr. Thomas Lentz, Manager, FENOC Nuclear Licensing & Regulatory Affairs, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 5, 2019.

Sincerely,



Douglas B. Huey

**Davis-Besse Nuclear Power Station, Unit No. 1**

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**Enclosure:**

**Evaluation of Proposed Changes**

**cc: NRC Region III Administrator  
NRC Resident Inspector  
NRR Project Manager  
Executive Director, Ohio Emergency Management Agency,  
State of Ohio (NRC Liaison)  
Utility Radiological Safety Board**

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Subject: Proposed License and Associated Technical Specification Changes for  
Permanently Defueled Condition

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1. License and Technical Specification Page Markups
2. Technical Specification Bases Page Markups (for information only)



## **1.0 SUMMARY DESCRIPTION**

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," FirstEnergy Nuclear Operating Company (FENOC) proposes an amendment to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specifications (TS), of RFOL No. NPF-3 for Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The proposed license amendment request (LAR) would revise the RFOL and the associated TS to the permanently defueled technical specifications (PDTS) consistent with the permanent cessation of power operation and permanent defueling of the reactor.

By letter dated April 25, 2018 (Reference 1), FENOC provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) of permanent cessation of power operations at DBNPS by May 31, 2020. After docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel. In support of this, the DBNPS RFOL and associated TS are being proposed for revision to reflect the planned permanent shutdown and defueled condition.

The proposed changes to the RFOL and TS are in accordance with 10 CFR 50.36(c)(1) through (c)(5). The proposed changes also include administrative changes to content format and revised page numbering. The TS Table of Contents is revised accordingly.

The current DBNPS TS contain limiting conditions for operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including safe storage and management of irradiated fuel. 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 TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. 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 TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety that addresses the reduced scope of postulated design basis accidents associated with a permanently defueled plant.

In the development of the proposed PDTS changes, FENOC reviewed the PDTS requirements from other plants that have permanently shutdown, primarily Crystal River Nuclear Plant, Unit 3 (Reference 2), San Onofre Nuclear Generating Station, Units 2 and 3 (Reference 3), Kewaunee (Reference 4), and Fort Calhoun Station Unit 1 (Reference 5).

This LAR provides a discussion, description, and technical evaluation of the proposed RFOL and TS changes, and information supporting a finding of no significant hazards consideration (NSHC).

### **Related Licensing Actions**

By letter dated October 22, 2018 (Reference 17), FENOC submitted a LAR proposing changes to the organization, staffing, and training requirements contained in TS Section 5.0, Administrative Controls that complements and supports this request.

The NRC is also currently reviewing the licensing actions listed below that are unrelated to this request:

- A proposed change to the fire protection program in License Condition 2.C(4) and the removal of associated Technical Specification 5.4.1.d was submitted for approval by letter dated December 16, 2015 (Accession No. ML15350A314) and supplemented by letters dated March 7, 2016, July 28, 2016, December 16, 2016, January 17, 2017, June 16, 2017, October 9, 2017, April 2, 2018, September 11, 2018, and November 20, 2018 (Accession Nos. ML16067A195, ML16210A422, ML16351A330, ML17017A504, ML17170A000, ML17284A190, ML18094A798, ML18254A073, and ML18324A677 respectively). License Condition 2.C(4) is proposed for deletion in its entirety as discussed in Section 2 of this request, and therefore, the licensing actions are unrelated.
- A proposed change to the support agreement in License Condition 3.B was submitted for approval by letter dated May 18, 2017 (Accession No. ML17138A381) and supplemented by letter dated August 23, 2018 (Accession No. ML18235A194). License Condition 3.B is proposed for deletion in its entirety as discussed in Section 2 of this request, and therefore, the licensing actions are unrelated.

## **2.0 DETAILED DESCRIPTION AND BASIS FOR THE CHANGES**

The proposed amendment would revise the DBNPS RFOL and associated TS for a permanently shutdown and defueled condition. To support the proposed changes, FENOC has evaluated the design basis accidents (DBAs) that will be applicable in a permanently shutdown and defueled condition. FENOC has also evaluated the General Design Criteria (GDC) with respect to compliance in the permanently shutdown and defueled condition. The DBA and GDC evaluations provide the framework and basis for the proposed changes.

### **Design Basis Accident Analyses Applicable to Proposed Change**

Chapter 15 of the DBNPS Updated Final Safety Analysis Report (UFSAR) contains the DBAs and transient scenarios applicable to DBNPS. 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 reactor coolant system (RCS). Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems that could affect the reactor core.

With the termination of reactor operations at DBNPS and the permanent removal of fuel from the reactor as certified in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the majority of the DBA scenarios postulated in the UFSAR will no longer be possible. During decommissioning the irradiated fuel will be stored in the spent fuel pool (SFP) or the dry fuel storage facility (DFSF) until it is shipped offsite in accordance with the schedules to be provided in the Post Shutdown Decommissioning Activities Report (PSDAR) and the Spent Fuel Management Plan. The RCS, steam system, and turbine generator are no longer in operation and have no function related to the safe storage and management of the spent nuclear fuel.

Chapter 15 of the UFSAR describes the safety analysis aspects of the plant that were evaluated to demonstrate that the plant could be operated safely and that radiological consequences from postulated accidents do not exceed regulatory requirements. The full spectrum of abnormal situations and accidents is divided into three classes in accordance with their anticipated frequency and their radiological consequences as follows:

- a. Class 1 – Events Leading to No Radioactivity Release at Exclusion Area Boundary
- b. Class 2 – Events Leading to Small to Moderate Radioactivity Release at Exclusion Area Boundary.
- c. Class 3 – Design Basis Accidents

A list of the Chapter 15 DBAs and whether the accident applies to a permanently defueled condition is provided in Table 2.1.

The UFSAR Chapter 15 DBA accident scenarios that remain credible in the permanently defueled condition, with fuel stored in the SFP, are external causes, waste gas decay tank rupture (WGDTR), and a fuel handling accident (FHA).

Table 2.1 – DBNPS Design Basis Accidents

| <b>UFSAR Section</b> | <b>Postulated Accident or Transient</b>   | <b>Permanently Defueled Applicability</b> |
|----------------------|---|---|
| 15.2                 | Class 1 - Events Leading to No Radioactive Release at Exclusion Area Boundary   |   |
| 15.2.1               | Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident)  | Not Applicable                            |
| 15.2.2               | Uncontrolled Control Rod Assembly Group Withdrawal at Power   | Not Applicable                            |
| 15.2.3               | Control Rod Assembly Misalignment (Stuck-Out, Stuck-In, or Dropped Control Rod Assembly)  | Not Applicable                            |
| 15.2.4               | Makeup and Purification System Malfunction  | Not Applicable                            |
| 15.2.5               | Loss of Forced Reactor Coolant Flow (Partial, Complete, and Single Reactor Coolant Pump Locked Rotor)   | Not Applicable                            |
| 15.2.6               | Startup of an Inactive Reactor Coolant Loop (Pump Startup Accident)   | Not Applicable                            |
| 15.2.7               | Loss of External Electrical Load and/or Turbine Trip  | Not Applicable                            |
| 15.2.8               | Loss of Normal Feedwater  | Not Applicable                            |
| 15.2.9               | Loss of all AC Power to Station Auxiliaries (Station Blackout)  | Not Applicable                            |
| 15.2.10              | Excessive Heat Removal Due to Feedwater System Malfunction  | Not Applicable                            |
| 15.2.11              | Excessive Load Increase   | Not Applicable                            |
| 15.2.12              | Anticipated Variations in the Reactivity of the Reactor   | Not Applicable                            |
| 15.2.13              | Failure of Regulating Instrumentation   | Not Applicable                            |
| 15.2.14              | External Causes   | Applicable                                |
| 15.3                 | Class 2 – Events Leading to Small to Moderate Radioactive Releases at Exclusion Area Boundary   |   |
| 15.3.1               | Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates Emergency Core Cooling   | Not Applicable                            |
| 15.3.2               | Minor Secondary System Pipe Break   | Not Applicable                            |
| 15.3.3               | Inadvertent Loading of a Fuel Assembly Into an Improper Position  | Not Applicable                            |
| 15.4                 | Class 3 – Design Basis Accidents  |   |
| 15.4.1               | Waste Gas Decay Tank Rupture  | Applicable                                |
| 15.4.2               | Steam Generator Tube Rupture  | Not Applicable                            |
| 15.4.3               | CRA Ejection Accident   | Not Applicable                            |
| 15.4.4               | Steam Line Break  | Not Applicable                            |
| 15.4.5               | Break in Instrument Lines or Lines from Primary System That Penetrate Containment   | Not Applicable                            |
| 15.4.6               | Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss-of-Coolant Accident) | Not Applicable                            |
| 15.4.7               | Fuel Handling Accident  | Applicable                                |
| 15.4.8               | Effects of Toxic Material Release on the Control Room   | Not Applicable <sup>1</sup>               |

1: UFSAR Section 15.4.8 states that toxic materials are not stored in volumes which would affect control room habitability.



In accordance with 10 CFR 50.2, "Definitions," safety-related systems, structures, and components (SSCs) are those relied on to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant pressure boundary;
2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or,
3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.43(a)(1) or 100.11.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to a plant in a permanently defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after the termination of reactor operations at DBNPS and the permanent removal of the fuel from the reactor vessel, and following 95 days of decay time after shutdown (as discussed below), none of the SSCs at DBNPS meet the definition of a safety-related SSC stated in 10 CFR 50.2 (with the exception of the passive spent fuel pool structure).

10 CFR 50.36, "Technical specifications," promulgates the regulatory requirements related to the content of technical specifications. As detailed in subsequent sections of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS. In a permanently defueled condition, the scope of equipment and parameters that must be included in the DBNPS TS is limited to those needed to address the remaining applicable design basis accidents so that the consequences of the accidents are maintained within acceptable limits. The applicable accidents are as follows:

### **External Causes**

Storms and earthquakes have been considered in the facility design. SSCs important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions. Potential changes to structures after the reactor is permanently shutdown and defueled are not expected to alter the event response.

### **Waste Gas Decay Tank Rupture**

Waste gas decay tanks are used in the radioactive waste disposal system to store radioactive gaseous waste from the station until such time that the radioactive decay renders the gas safe for release to the site environment. Rupture of a waste gas decay tank would result in the premature release of its radioactive contents to the environment (UFSAR 15.4.1). The WGDTR accident remains valid after permanent defueling.

Following permanent shutdown, the waste gas tanks will be required to retain and release waste gas generated from water management activities. The existing WGDTR accident (UFSAR 15.4.1) assumes that the waste gas tank contains gaseous activity evolved from the RCS following operation with up to one percent defective fuel. The tank ruptures and releases its contents to the auxiliary building, which is then vented to the atmosphere through charcoal filters over a two-hour period.

The new FHA after permanent shutdown, "Radiological Consequences of a Fuel Handling Accident Outside Containment After Permanent Shutdown," demonstrates that the unmitigated release of 651.1 Curies (Ci) of dose-equivalent krypton-85 (whole body and skin) and 5.56 Ci of dose-equivalent iodine-131 (thyroid) does not exceed the dose limits for the control room (CR), exclusion area boundary (EAB), or low population zone (LPZ). Since an atmospheric dispersion factor ( $X/Q$ ) of one second per cubic meter ( $\text{sec}/\text{m}^3$ ) was used for the CR in the new FHA analysis, this bounds any release path from a possible WGDTR. Likewise, the new FHA uses the existing  $X/Q$  for the EAB and LPZ, and therefore, these are also still applicable for the WGDTR. Thus, if the activity in the waste gas decay tank is less than the maximum activity assumed to be released in the FHA analysis after permanent shutdown, then the resultant doses due to a rupture will remain within established limits. Since the results are based on the new FHA analysis, there is no credit for emergency ventilation filtration, CR isolation, the CR volume, or CR filtration either. Prior to implementing the PDTs, the concentrations in the waste gas decay tanks will be measured and verified to be less than 651.1 Ci of dose-equivalent krypton-85 and 5.56 Ci of dose-equivalent iodine-131.

Measuring and verifying the activity in the waste gas decay tanks would only be performed once, because upon permanent shutdown and cooldown, the source term contained within the waste gas decay tanks represents the highest (worst case) source term and is expected to be significantly less than that assumed in the WGDTR analysis. Subsequent additions to the waste gas decay tank resulting from water management activities would be less than the final shutdown and cooldown waste gas tank source term.

### **Fuel Handling Accident Analysis for the Permanently Defueled Condition**

A new FHA analysis, "Radiological Consequences of a Fuel Handling Accident Outside Containment After Permanent Shutdown," in the SFP for the permanently defueled condition has been completed. This post-permanent shutdown FHA was evaluated using the assumptions and methodology described in Regulatory Guide 1.25 (March 1972). The new analysis does not credit the function of any active mitigation measures.

Consistent with the current licensing basis, the FHA is defined as the dropping of a single spent fuel assembly in the SFP during fuel handling activities, such that the entire outer row of fuel rods in the assembly, 56 of 208, suffers mechanical damage to the cladding. This accident is postulated to occur despite the administrative controls and physical limitations imposed on fuel-handling operations. The gap activity in the damaged rods is



instantaneously released into the SFP. The release occurs under 23 feet of water, and a decontamination factor for iodine is used consistent with Regulatory Guide 1.25.

Once the reactor has been permanently defueled, the new FHA prohibits recently-irradiated fuel movement until the sources have decayed adequately. That is, handling of irradiated fuel that has occupied part of a critical reactor core within the previous 95 days is not permitted. The source term is taken from the licensing basis FHA analysis and adjusted for the additional decay time. Similarly, the release fractions and pool water decontamination factor are also taken from the licensing basis FHA analysis and are consistent with Regulatory Guide 1.25. The new FHA analysis uses a X/Q of 1 sec/m<sup>3</sup> to conservatively bound any meteorological data, release points, and CR receptor points. This effectively assumes that there is no dispersion and the accident occurs at the CR. Dose to the CR operators is not dependent on the accident release time, as the analysis assumes a simplified puff release that instantaneously transports the released activity to the CR. No credit is taken for emergency ventilation filtration, CR isolation, the CR volume, or CR filtration.

The new FHA uses the existing X/Q for the EAB and LPZ.

Consistent with the licensing basis FHA evaluation, the acceptance criteria for CR doses are taken as 5 rem whole body, 30 rem thyroid, and 30 rem skin. These are based on 10 CFR 50 Appendix A, GDC 19, and Chapter 6.4 of NUREG-0800. The EAB and LPZ acceptance criteria of 6 rem whole body and 75 rem thyroid are from Chapter 15.7.4 of NUREG-0800.

Without crediting mitigation by any active SSC, the dose consequences of the new FHA at 95 days after reactor shutdown is as follows:

| Location | Dose Limits                                       | Dose Analysis Results   |
|----------|---|---|
| CR       | 5 rem whole body<br>30 rem skin<br>30 rem thyroid | 0.35 rem whole body<br>27.67 rem skin<br>20.84 rem thyroid                |
| EAB      | 6 rem whole body<br>75 rem thyroid                | 6.58x10 <sup>-5</sup> rem whole body<br>3.96x10 <sup>-3</sup> rem thyroid |
| LPZ      | 6 rem whole body<br>75 rem thyroid                | 3.43x10 <sup>-6</sup> rem whole body<br>2.06x10 <sup>-4</sup> rem thyroid |

In conclusion, the FHA analysis for DBNPS shows that, following 95 days of decay time after reactor shutdown and provided the SFP water level requirements of TS 3.7.14 are met<sup>1</sup>, the dose consequences for the CR, the EAB and the LPZ remain below the

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<sup>1</sup> TS 3.7.14, "Spent Fuel Pool Water Level," requires the spent fuel pool water level to be greater than or equal to 23 feet over the top of irradiated fuel assemblies seated in the storage racks. TS 3.7.14 is applicable during movement of irradiated fuel assemblies in the spent fuel pool.

acceptance criteria, without relying on active components remaining functional for accident mitigation during and following the event.

Once FENOC docketed the permanent cessation of power operations and permanent removal of fuel from the reactor vessel, it is desirable to implement the proposed LAR as soon as possible to support the decommissioning schedule. Therefore, to preclude the movement of fuel before the 95-day FHA accident input assumption, the following new license condition 2.I is proposed:

“2.I Handling of irradiated fuel that has occupied part of a critical reactor core within the previous 95 days is not permitted.”

The new licensing condition is discussed in the detailed license changes later in this submittal.

### **DBA Conclusion**

The remaining DBAs that support permanently shutdown and defueled condition do not rely on any active safety system for mitigation. The new FHA analysis, after 95 days of the permanent shutdown, demonstrates that the unmitigated release of 651.1 Ci of dose-equivalent krypton-85 (whole body and skin) and 5.56 Ci of dose-equivalent iodine-131 (thyroid) will not exceed the limits for the CR, EAB or LPZ. The activity within the waste gas tanks will be confirmed to less the 651.1 Ci of dose-equivalent krypton-85 and 5.56 Ci of dose-equivalent iodine-131 prior to implementation of the PDTs. Upon permanent shutdown and cooldown, the source term contained within the waste gas decay tanks represents the highest (worst case) source term and is expected to be significantly less than that assumed in the WGDTR analysis. Subsequent additions to the waste gas decay tanks resulting from water management activities would be less than the final shutdown and cooldown waste gas tank source term.

### **Detailed Review of General Design Criteria After Permanent Defueling**

The GDC became effective after the DBNPS construction permit was issued. SECY-92-223, dated September 18, 1992 (Accession No. ML003763736) summarized the results of a Commission vote in which the Commissioners instructed the NRC staff not to apply the GDC to plants with construction permits issued prior to May 21, 1971. The DBNPS construction permit was issued on March 24, 1971. However, DBNPS UFSAR Appendix 3D describes how DBNPS meets the intent of the GDC in 10 CFR 50 Appendix A. With the termination of reactor operations at DBNPS and the permanent removal of fuel from the reactor as certified in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the majority of the GDC in the UFSAR will no longer be applicable. During decommissioning, the irradiated fuel will be stored in the SFP or in the DFSF until it is shipped offsite in accordance with the schedules to be provided in the PSDAR and the Spent Fuel Management Plan. The RCS, steam system, and turbine generator are no longer in operation and have no function related to the safe

storage and management of the spent nuclear fuel. In general, the GDC that relate only to reactor operation or the systems that support reactor operation will no longer be applicable when the facility is in a permanently defueled condition. However, since fuel and radioactive waste will still be stored at the facility, the GDC that relate to the storage of waste, fuel, and the prevention of radioactive release will still be applicable to the facility. This includes supporting GDCs that relate to quality standards and fire protection.

Compliance with each of the 10 CFR 50 Appendix A GDC was reviewed as applied to the permanently shutdown and defueled condition and the limitations imposed by 10 CFR 50.82(a)(2) upon docketing the certification required by 10 CFR 50.82(a)(1). This review determined that compliance could be expressed in four categories as follows:

- A. No Longer Applies – Compliance with the GDC is no longer applicable to DBNPS since the intent and scope are based on conditions that do not apply to the facility in a permanently shutdown and defueled condition. The DBAs that evaluate conditions applicable to operation of the reactor no longer apply. The DBAs that are applicable to DBNPS in a permanently shutdown and defueled condition do not credit active safety systems for accident mitigation.
- B. Unchanged – Compliance with the GDC continues to apply to DBNPS as described in UFSAR Chapter 3, Appendix D. The scope and intent of the GDC is not impacted by the transition from operating status to permanently shutdown and defueled status.
- C. Minor Change – Compliance with the GDC is still required for DBNPS; however, the scope can be reduced based on the transition from operating status to permanently shutdown and defueled status.
- D. Major Change – Compliance with the GDC is still required for DBNPS; however, the intent and scope are impacted by the transition from operating status to permanently shutdown and defueled status. These criteria are discussed in further detail below, reflecting the proposed changes to the DBNPS licensing basis.

A list of the current 10 CFR 50 Appendix A GDC, and their applicability to DBNPS in a permanently shutdown and defueled condition, are provided in Table 2.2.

Table 2.2 – Compliance with 10 CFR Appendix A GDC

| <b>10 CFR 50 Appendix A General Design Criteria</b>                               | <b>Applicability to Permanently Defueled DBNPS</b> |
|---|--|
| Criterion 1 - Quality Standards and Records                                       | B. Unchanged                                       |
| Criterion 2 - Design Bases for Protection Against Natural Phenomena               | C. Minor Change                                    |
| Criterion 3 - Fire Protection   | D. Major Change                                    |
| Criterion 4 - Environmental and Dynamic Effects Design Bases                      | A. No Longer Applies                               |
| Criterion 5 - Sharing of Structures, Systems, and Components                      | A. No Longer Applies                               |
| Criterion 10 - Reactor Design   | A. No Longer Applies                               |
| Criterion 11 - Reactor inherent Protection  | A. No Longer Applies                               |
| Criterion 12 - Suppression of Reactor Power Oscillations                          | A. No Longer Applies                               |
| Criterion 13 - Instrumentation and Control  | A. No Longer Applies                               |
| Criterion 14 - Reactor Coolant Pressure Boundary                                  | A. No Longer Applies                               |
| Criterion 15 - Reactor Coolant System Design                                      | A. No Longer Applies                               |
| Criterion 16 - Containment Design   | A. No Longer Applies                               |
| Criterion 17 - Electric Power Systems   | A. No Longer Applies                               |
| Criterion 18 - Inspection and Testing of Electric Power Systems                   | A. No Longer Applies                               |
| Criterion 19 - Control Room   | A. No Longer Applies                               |
| Criterion 20 - Protection System Functions  | A. No Longer Applies                               |
| Criterion 21 - Protection System Reliability and Testability                      | A. No Longer Applies                               |
| Criterion 22 - Protection System Independence                                     | A. No Longer Applies                               |
| Criterion 23 - Protection System Failure Modes                                    | A. No Longer Applies                               |
| Criterion 24 - Separation of Protection and Control Systems                       | A. No Longer Applies                               |
| Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions | A. No Longer Applies                               |
| Criterion 26 - Reactivity Control System Redundancy and Capability                | A. No Longer Applies                               |
| Criterion 27 - Combined Reactivity Control Systems Capability                     | A. No Longer Applies                               |
| Criterion 28 - Reactivity Limits  | A. No Longer Applies                               |
| Criterion 29 - Protection Against Anticipated Operational Occurrences             | A. No Longer Applies                               |
| Criterion 30 - Quality of Reactor Coolant Pressure Boundary                       | A. No Longer Applies                               |
| Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary           | A. No Longer Applies                               |
| Criterion 32 - Inspection of Reactor Coolant Pressure Boundary                    | A. No Longer Applies                               |
| Criterion 33 - Reactor Coolant Makeup   | A. No Longer Applies                               |



| <b>10 CFR 50 Appendix A General Design Criteria</b>                            | <b>Applicability to Permanently Defueled DBNPS</b> |
|--|--|
| Criterion 34 - Residual Heat Removal   | A. No Longer Applies                               |
| Criterion 35 - Emergency Core Cooling  | A. No Longer Applies                               |
| Criterion 36 - Inspection of Emergency Core Cooling System                     | A. No Longer Applies                               |
| Criterion 37 - Testing of Emergency Core Cooling System                        | A. No Longer Applies                               |
| Criterion 38 - Containment Heat Removal  | A. No Longer Applies                               |
| Criterion 39 - Inspection of Containment Heat Removal System                   | A. No Longer Applies                               |
| Criterion 40 - Testing of Containment Heat Removal System                      | A. No Longer Applies                               |
| Criterion 41 - Containment Atmosphere Cleanup                                  | A. No Longer Applies                               |
| Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems            | A. No Longer Applies                               |
| Criterion 43 - Testing of Containment Atmosphere Cleanup Systems               | A. No Longer Applies                               |
| Criterion 44 - Cooling Water   | A. No Longer Applies                               |
| Criterion 45 - Inspection of Cooling Water System                              | A. No Longer Applies                               |
| Criterion 46 - Testing of Cooling Water System                                 | A. No Longer Applies                               |
| Criterion 50 - Containment Design Basis  | A. No Longer Applies                               |
| Criterion 51 - Fracture Prevention of Containment Pressure Boundary            | A. No Longer Applies                               |
| Criterion 52 - Capability for Containment Leakage Rate Testing                 | A. No Longer Applies                               |
| Criterion 53 - Provisions for Containment Testing and Inspection               | A. No Longer Applies                               |
| Criterion 54 - Piping Systems Penetrating Containment                          | A. No Longer Applies                               |
| Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment       | A. No Longer Applies                               |
| Criterion 56 - Primary Containment Isolation                                   | A. No Longer Applies                               |
| Criterion 57 - Closed Systems Isolation Valves                                 | A. No Longer Applies                               |
| Criterion 60 - Control of Releases of Radioactive Materials to the Environment | B. Unchanged                                       |
| Criterion 61 - Fuel Storage and Handling and Radioactivity Control             | B. Unchanged                                       |
| Criterion 62 - Prevention of Criticality in Fuel Storage and Handling          | B. Unchanged                                       |
| Criterion 63 - Monitoring Fuel and Waste Storage                               | B. Unchanged                                       |
| Criterion 64 - Monitoring Radioactivity Releases                               | C. Minor Change                                    |

### Criterion 3 – Fire Protection

Structures, systems, and components important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials are used wherever practical throughout the unit, particularly in locations such as the containment, control room and areas containing components of engineered safety features. Fire-detection and fighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

#### *Discussion:*

Regulatory Guide 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," describes the changes of the fire protection program related to operating unit as required by GDC 3 transitioning to a permanently shutdown condition.

The primary objectives of the fire protection program for operating reactors are to minimize fire damage to structures, systems, and components (SSCs) important to safety; to ensure the capability to safely shut down the reactor; and to maintain it in a safe shutdown condition. For an initial period following shutdown, accidents that can challenge the 10 CFR Part 100 limits remain credible. The fire protection program should continue to provide protection against these events. The primary fire protection concern for permanently shutdown plants is protecting the integrity of the spent fuel and preventing or minimizing the release of radioactive materials resulting from fires involving contaminated plant SSCs or radioactive wastes. The radiation dose limits specified in 10 CFR Part 20, "Standards for Protection Against Radiation," apply to plant personnel and members of the public for fire incidents at permanently shutdown nuclear power plants. Licensees should make every effort to maintain exposures to radiation resulting from a fire as low as reasonably achievable.

The fire protection program for a decommissioned unit is governed by the requirements of 10 CFR 50.48(f).

### Detailed Discussion of Proposed RFOL and TS Changes

The existing DBNPS TS contain 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 functions related to the safe storage and management of irradiated fuel at an operating plant is similar to the corresponding

function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. The proposed license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certification of cessation of power and permanent removal of fuel from the reactor vessel is docketed. Therefore, license conditions contained in the RFOL, the LCOs and associated SRs that do not apply in a defueled condition are being proposed for deletion. Additionally, several of the license conditions in the RFOL are being proposed for revision. The incorporation of the changes will result in the PDTs that will continue to provide a commensurate level of safety that addresses the reduced scope of postulated DBAs associated with a permanently defueled plant.

The following tables identify each RFOL and TS section that is being changed, the proposed change, and the basis for each change. Changes to the RFOL are addressed first, followed by the TS. Proposed revisions are shown in ***Bold-Italics*** and deletions are shown using *italicized strikethrough*.

Attachment 1 provides the marked-up version of the DBNPS RFOL and TS. The TS that are deleted in their entirety are identified as such below, but the associated deleted pages are not included in Attachment 1. Proposed changes to the TS Bases addressing the proposed changes to the relevant TS are provided for information only in Attachment 2. Upon approval of this amendment, changes to the TS Bases will be incorporated in accordance with TS 5.5.13, "Technical Specifications (TS) Bases Control Program," which is retained in its entirety without change.

In addition, the proposed changes to the TS are considered a major rewrite. Revised formatting (margins, font, tabs, and so forth) of content is used to create a continuous electronic file, revised numbering of sections and pages; and the deletion of unused placeholders, where appropriate, is used to condense and reduce the number of pages in the TS without affecting the technical content. Since the changes to the TS are considered a major rewrite, revision bars are not used. The TS Table of Contents is revised to reflect the remaining applicable sections and new page numbering. These changes are considered administrative and are shown in the marked-up pages (Attachment 1).

10 CFR 50.36, "Technical specifications," promulgates the regulatory requirements related to the content of TS. As detailed in subsequent sections of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS. In a permanently defueled condition, the scope of equipment and parameters that must be included in the DBNPS TS is limited to those needed to address the remaining applicable design basis accidents (the postulated FHA and WGDTR) so that the consequences of the accidents are maintained within acceptable limits.

### RENEWED FACILITY ~~OPERATING~~ LICENSE

The title of this section will be revised to remove the term “Operating.” Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “Operating” description provides accuracy in the 10 CFR Part 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.

#### License Condition 1.A. Footnote on Page L-1

##### Current Footnote

FENOC is authorized to act as agent for FirstEnergy Nuclear Generation, LLC, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

##### Proposed Footnote

FENOC is authorized to act as agent for FirstEnergy Nuclear Generation, LLC, and has exclusive responsibility and control over the physical construction, ~~operation~~, and maintenance of the facility.

#### **Basis**

The proposed change to delete the term “operation” provides a more accurate description of the future license. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “operation” description provides accuracy in the 10 CFR Part 50 license description.

#### License Condition 1.B.

##### Current License Condition 1.B.

Construction of the Davis-Besse Nuclear Power Station, Unit No. 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-80 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

##### Proposed License Condition 1.B.

***Deleted per Amendment No. ###.***



| Basis  |
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| This license condition will be deleted in its entirety. Decommissioning of DBNPS is not dependent on the regulations that govern construction of the facility. |

| License Condition 1.C.  |   |
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| <u>Current License Condition 1.C.</u><br><br>The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;  | <u>Proposed License Condition 1.C.</u><br><br>The facility will <del>operate</del> <b>be maintained</b> in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; |
| Basis   |   |
| The proposed change to the description “The facility will operate” to the facility “will be maintained” provides a more accurate description of the future requirements. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “operating” description provides accuracy in the 10 CFR Part 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant. |   |

| License Condition 1.D.   |  |
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| <u>Current License Condition 1.D.</u><br><br>There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; | <u>Proposed License Condition 1.D.</u><br><br>There is reasonable assurance: (i) that the activities authorized by this renewed <del>operating</del> 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 rules and regulations of the Commission; |
| Basis  |  |
| The proposed change to delete the term “operating” provides a more accurate description of the future license. Once FENOC docket the certifications required by  |  |

10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “operating” description provides accuracy in the 10 CFR Part 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.

| License Condition 1.E.   |  |
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| <u>Current License Condition 1.E.</u><br><br>The FirstEnergy Nuclear Operating Company is technically qualified and the licensees are financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;   | <u>Proposed License Condition 1.E.</u><br><br>The FirstEnergy Nuclear Operating Company is technically qualified and the licensees are financially qualified to engage in the activities authorized by this renewed <del>operating</del> license in accordance with the rules and regulations of the Commission; |
| Basis  |  |
| The proposed change to delete the term “operating” provides a more accurate description of the future license. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “operating” description provides accuracy in the 10 CFR 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant. |  |

| License Condition 1.G.   |  |
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| <u>Current License Condition 1.G.</u><br><br>The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public; | <u>Proposed License Condition 1.G.</u><br><br>The issuance of this renewed <del>operating</del> license will not be inimical to the common defense and security or to the health and safety of the public; |
| Basis  |  |
| The proposed change to delete the term “operating” provides a more accurate description of the future license. Once FENOC docket the certifications required by                                |  |

10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “operating” description provides accuracy in the 10 CFR 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.

| <b>License Condition 1.H.</b>   |  |
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| <p><u>Current License Condition 1.H.</u></p> <p>After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility Operating License No. NPF-3 subject to the conditions for protection of the environment set forth herein is in accordance with 10 CFR Part 51 (formerly Appendix D to 10 CFR Part 50), of the Commission’s regulations and all applicable requirements have been satisfied;</p>  | <p><u>Proposed License Condition 1.H.</u></p> <p>After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility <del>Operating</del> License No. NPF-3 subject to the conditions for protection of the environment set forth herein is in accordance with 10 CFR Part 51 (formerly Appendix D to 10 CFR Part 50), of the Commission’s regulations and all applicable requirements have been satisfied;</p> |
| <b>Basis</b>  |  |
| <p>The proposed change to delete the term “Operating” from the title of the facility, provides a more accurate description of the future license. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “Operating” description provides accuracy in the 10 CFR 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.</p> |  |

| <b>License Condition 1.I.</b>   |   |
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| <p><u>Current License Condition 1.I.</u></p> <p>The receipt, possession, and use of source, byproduct and special nuclear</p> | <p><u>Proposed License Condition 1.I.</u></p> <p><b><i>Deleted per Amendment No. ###.</i></b></p> |

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| material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Part 30, 40, and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31; and   |  |
| <b>Basis</b>  |  |
| This license condition is proposed for deletion in its entirety. The Commission's finding regarding possession and use of byproduct, source, and special nuclear material is not dependent on decommissioning of the facility. Additionally, possession and use of byproduct, source, and special nuclear material at DBNPS during decommissioning activities is covered by License Condition 2.B.(4), which will remain in effect. Therefore, License Condition 1.I is not needed. |  |

| <b>License Condition 1.J.</b>   |   |
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| <u>Current License Condition 1.J.</u><br><br>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 the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations. | <u>Proposed License Condition 1.J.</u><br><br>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 <del>operation</del> <b>facility maintenance</b> 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 the renewed <del>operating</del> -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. |
| <b>Basis</b>  |   |



The proposed change to delete the term “operating” and replacement of “operation” for “facility maintenance,” provides a more accurate description of the future license. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “operating” description provides accuracy in the 10 CFR 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.

| License Condition 2.   |  |
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| <u>Current License Condition 2.</u><br><br>Renewed Facility Operating License No. NPF-3 is hereby issued to FirstEnergy Nuclear Operating Company (FENOC), and FirstEnergy Nuclear Generation, LLC to read as follows:   | <u>Proposed License Condition 2.</u><br><br>Renewed Facility <del>Operating</del> License No. NPF-3 is hereby issued to FirstEnergy Nuclear Operating Company (FENOC), and FirstEnergy Nuclear Generation, LLC to read as follows: |
| Basis  |  |
| The proposed change to delete the term “Operating” from the title of the facility, provides a more accurate description of the future license. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the “Operating” description provides accuracy in the 10 CFR 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant. |  |

| License Condition 2.A.   |  |
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| <u>Current License Condition 2.A.</u><br><br>This renewed license applies to the Davis-Besse Nuclear Power Station, Unit No. 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by FirstEnergy Nuclear Generation, LLC. The facility is located on the south-western shore of Lake Erie in Ottawa County, Ohio, | <u>Proposed License Condition 2.A.</u><br><br>This renewed license applies to the Davis-Besse Nuclear Power Station, Unit No. 1, a <b>permanently defueled</b> pressurized water nuclear reactor and associated equipment (the facility), owned by FirstEnergy Nuclear Generation, LLC. The facility is located on the south-western |

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| approximately 21 miles east of Toledo, Ohio, and is described in the “Final Safety Analysis Report” as supplemented and amended (Amendments 14 through 44) and the Environmental Report as supplemented and amended (Supplements 1 through 2). | shore of Lake Erie in Ottawa County, Ohio, approximately 21 miles east of Toledo, Ohio, and is described in the “Final Safety Analysis Report” as supplemented and amended (Amendments 14 through 44) and the Environmental Report as supplemented and amended (Supplements 1 through 2). |
| <b>Basis</b>   |   |
| The description of the unit is updated to reflect the permanently defueled status of the facility.   |   |

| <b>License Condition 2.B.(1)</b>   |  |
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| <u>Current License Condition 2.B.(1)</u><br><br>FENOC, pursuant to Section 103 of the Act and 10 CFR Part 50, “Licensing of Production and Utilization Facilities,” to possess, use, and operate the facility;   | <u>Proposed License Condition 2.B.(1)</u><br><br>FENOC, pursuant to Section 103 of the Act and 10 CFR Part 50, “Licensing of Production and Utilization Facilities,” to possess, <del>use, and operate</del> <b>and use</b> the facility <b>as required for nuclear fuel storage</b> ; |
| <b>Basis</b>   |  |
| The proposed language change associated with operating the facility in this license condition is removed. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The proposed license change will allow use of the facility as required for nuclear fuel storage. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled reactor vessel. |  |

| <b>License Condition 2.B.(3)</b>  |   |
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| <u>Current License Condition 2.B.(3)</u><br><br>FENOC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any | <u>Proposed License Condition 2.B.(3)</u><br><br>FENOC, pursuant to the Act and 10 CFR Part 70, to <del>receive</del> , possess <del>and use</del> at |

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| 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 Safety Analysis Report, as supplemented and amended;   | any time special nuclear material <b><i>that was used</i></b> as reactor fuel, in accordance with the limitations for storage <del>and amounts required for reactor operation</del> , as described in the Final Safety Analysis Report, as supplemented and amended; |
| <b>Basis</b>   |  |
| <p>The language in this license condition is proposed to be changed to reflect that special nuclear material for fuel used for reactor operations can no longer be received, only stored. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.</p> |  |

| <b>License Condition 2.B.(4)</b>   |   |
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| <p><u>Current License Condition 2.B.(4)</u></p> <p>FENOC, 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;</p>  | <p><u>Proposed License Condition 2.B.(4)</u></p> <p>FENOC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess and use at any time any byproduct, source <del>and special nuclear material as sealed neutron sources for reactor startup</del>, or sealed sources for <del>reactor instrumentation</del> <del>and</del> radiation monitoring equipment calibration, <del>and as fission detectors in amounts as required</del> <b>and to possess any byproduct, source and special nuclear material as sealed neutron sources previously used for reactor startup and reactor instrumentation; and fission detectors;</b></p> |
| <b>Basis</b>   |   |
| <p>The requirements regarding receipt of sealed neutron sources for reactor startup and nuclear instrumentation is proposed for deletion. This license condition is revised to reflect authorization only for continued possession of those sources used for reactor startups, produced as a byproduct, and those required for calibration. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the DBNPS license will no</p> |   |

longer authorize use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR 50.82(a)(2); therefore, the use of startup sources will no longer be needed. Therefore, the changes are consistent with the requirements associated with the decommissioning plant. The use of sources for radiation monitoring will continue to be required.

| License Condition 2.B.(6)  |   |
|--|---|
| <u>Current License Condition 2.B.(6)</u><br><br>FENOC, 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.   | <u>Proposed License Condition 2.B.(6)</u><br><br>FENOC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials <del>as may be</del> <b>that were</b> produced by the operation of the facility. |
| Basis  |   |
| This license condition will be revised to replace “as may be” for “that were.” This license condition is proposed for revision to allow possession of byproduct and special nuclear materials that were produced during operation of the reactor but not allow the separation of material that was produced by operations of the reactor. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant. |   |

| License Condition 2.C(1)  |  |
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| <u>Current License Condition 2.C(1)</u><br><br><u>Maximum Power Level</u><br><br>FENOC is authorized to operate the facility at steady state reactor core power levels not in excess of 2817 megawatts (thermal). Prior to attaining the power level, Toledo Edison Company shall comply with the conditions identified in Paragraph (3) (o) below and complete the preoperational tests, startup tests and other items | <u>Proposed License Condition 2.C(1)</u><br><br><b><i>Deleted per Amendment No. ###.</i></b> |



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| identified in Attachment 2 to this license in the sequence specified. Attachment 2 is an integral part of this renewed license.   |  |
| <b>Basis</b>  |  |
| <p>The requirements associated with the plant's maximum power level are proposed for deletion, since DBNPS will permanently be ceasing power operations. Since the DBNPS license will no longer allow the use of the facility for power operation as provided in 10 CFR 50.82(a)(2), the use of a power limit is no longer needed. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.</p> |  |

| <b>License Condition 2.C(2)</b>  |  |
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| <u>Current License Condition 2.C(2)</u><br><br><u>Technical Specifications</u><br><br><p>The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby incorporated in the renewed license. FENOC shall operate the facility in accordance with the Technical Specifications.</p>   | <u>Proposed License Condition 2.C(2)</u><br><br><u>Technical Specifications</u><br><br><p>The Technical Specifications contained in Appendix A, as revised through Amendment No. <del>297###</del>, are hereby incorporated in the renewed license. FENOC shall <del>operate</del> <b>maintain</b> the facility in accordance with the <b>Permanently Defueled</b> Technical Specifications.</p> |
| <b>Basis</b>   |  |
| <p>The proposed change incorporates the PDTs. Also changed is the designation from operating to maintaining the facility, which describes the defueled condition in which the DBNPS license will no longer allow the use of the facility for power operation as provided in 10 CFR 50.82(a)(2). Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.</p> |  |

| <b>License Condition 2.C(3)</b>   |   |
|---|---|
| <u>Current License Condition 2.C(3)</u><br><br><u>Additional Conditions</u> | <u>Proposed License Condition 2.C(3)</u><br><br><b>Deleted per Amendment No. ###.</b> |

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

- (a) FENOC shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation.
- (b) Deleted per Amendment 6
- (c) Deleted per Amendment 5
- (d) Prior to operation beyond 32 Effective Full Power Years, FENOC shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.
- (e) Deleted per Amendment 33
- (f) Deleted per Amendment 33
- (g) Deleted per Amendment 33
- (h) Deleted per Amendment 24
- (i) Deleted per Amendment 11
- (j) Revised per Amendment 3  
Deleted per Amendment 28
- (k) Within 60 days of startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall complete tests and obtain test results as required by the Commission to verify that faults on non-Class IE circuits would not propagate to the Class IE circuits in the Reactor Protection System and the Engineered Safety Features Actuation System.

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| <ul style="list-style-type: none"> <li>(l) Revised per Amendment 7<br/>Deleted per Amendment 15</li> <li>(m) Deleted per Amendment 7</li> <li>(n) Deleted per Amendment 10</li> <li>(o) Deleted per Amendment 2</li> <li>(p) Deleted per Amendment 29</li> <li>(q) Deleted per Amendment 7</li> <li>(r) Deleted per Amendment 30</li> <li>(s) Toledo Edison Company shall be exempted from the requirements of Technical Specification 3/4.7.8.1 for the two (2) Americium-Beryllium-Copper startup sources to be installed or already installed for use during the first refueling cycle until such time as the sources are replaced.</li> <li>(t) Added per Amendment 83<br/>Deleted per Amendment 122</li> </ul>   |  |
| <b>Basis</b>  |  |
| <p>This license condition is proposed for deletion in its entirety. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the DBNPS license will no longer authorize use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR 50.82(a)(2). This license condition references operational MODES 1 and 2, submittal of analysis to ensure protection against low temperature RCS overpressure events, completion of tests already completed, and startup sources. These conditions will no longer be applicable in the permanently defueled condition; therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant.</p> |  |

| <b>License Condition 2.C(4)</b>   |   |
|---|---|
| <p><u>Current License Condition 2.C(4)</u></p> <p><u>Fire Protection</u></p> <p>FENOC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report and as approved in the SERs dated July 26, 1979,</p> | <p><u>Proposed License Condition 2.C(4)</u></p> <p><b><i>Deleted per Amendment No. ###.</i></b></p> |

| <p>and May 30, 1991, subject to the following provision:</p> <p>FENOC may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.</p>  |  |
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| Basis   |  |
| <p>The proposed change deletes this license condition. This license condition, which is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48 with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire, is no longer applicable at DBNPS. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant 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. 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 shutdown and defueled plant is not required.</p> <p>As discussed in Section 1 of this LAR, the NRC is currently reviewing a licensing action for this condition. As License Condition 2.C(4) is proposed for deletion in its entirety with this request, the licensing actions are unrelated.</p> |  |

| License Condition 2.C(6)   |   |
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| <p><u>Current License Condition 2.C(6)</u></p> <p><u>Antitrust Conditions</u></p> <p>FENOC and FirstEnergy Nuclear Generation, LLC shall comply with the antitrust conditions delineated in Condition 2.E of this renewed license as if named therein. FENOC shall not market or broker power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1. FirstEnergy Nuclear Generation, LLC is responsible and accountable for the actions</p> | <p><u>Proposed License Condition 2.C(6)</u></p> <p><b><i>Deleted per Amendment No. ###.</i></b></p> |

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| of FENOC to the extent that said actions affect the marketing and brokering of power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1, and in any way, contravene the antitrust license conditions contained in the renewed license.  |  |
| <b>Basis</b>  |  |
| The requirements imposed to address the antitrust concerns is proposed for deletion. This license condition was imposed to address antitrust concerns associated with the operation of DBNPS. Since the DBNPS license no longer allows the use of the facility for power operation as provided in 10 CFR 50.82(a)(2), the antitrust conditions are no longer needed. Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant. |  |

| <b>License Condition 2.C(7)</b>  |   |
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| <p><u>Current License Condition 2.C(7)</u></p> <p><u>Steam Generator Tube Circumferential Crack Report</u></p> <p>Following each inservice inspection of steam generator tubes, the NRC shall be notified by FENOC of the following prior to returning the steam generators to service:</p> <ul style="list-style-type: none"> <li>a. Indications of circumferential cracking inboard of the roll repair.</li> <li>b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no reroll is present.</li> <li>c. Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.</li> </ul> | <p><u>Proposed License Condition 2.C(7)</u></p> <p><b><i>Deleted per Amendment No. ###.</i></b></p> |



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| <p>FENOC shall demonstrate by evaluation that the primary-to-secondary leakage following a LBLOCA, if any, as described in Appendix A to topic Report BAW-2374, July 2000, continues to be acceptable, based on the as-found condition of the steam generators. For the purpose of this evaluation, acceptable means that a best estimate of the leakage expected in the event of a LBLOCA would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR Part 100 limits). This is required to demonstrate that adequate margin and defense-in-depth continue to be maintained. A written summary of this evaluation shall be provided to the NRC within three months following completion of the steam generator tube inservice inspection.</p> |  |
| <b>Basis</b>   |  |
| <p>The requirements associated with steam generator tube circumferential cracking report are proposed for deletion. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2); therefore, the steam generators will no longer be used. As a result, a written summary evaluation to demonstrate that the primary to secondary leakage following a large break loss of coolant accident (LBLOCA) continues to be acceptable will no longer be required. Therefore, a license condition requiring such an evaluation for a permanently shutdown and defueled plant is not required.</p>              |  |

| <b>License Condition 2.C(9)</b>  |   |
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| <p><u>Current License Condition 2.C(9)</u></p> <p><u>Implementation of New and Revised Surveillance Requirements</u></p> <p>For SRs that are new in Amendment No. 279, the first performance is due at the end of the first surveillance interval, which</p> | <p><u>Proposed License Condition 2.C(9)</u></p> <p><b><i>Deleted per Amendment No. ###.</i></b></p> |

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| <p>begins on the date of implementation of this amendment.</p> <p>For SRs that existed prior to Amendment No. 279, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.</p> <p>For SRs that existed prior to Amendment No. 279, that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.</p> <p>For SRs that existed prior to Amendment No. 279, whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.</p> |  |
| <b>Basis</b>  |  |
| <p>The implementation of new and revised surveillance requirements provided in Amendment 279 is proposed for deletion. The implementation of these requirements was completed in accordance with the schedule specified by the NRC on the issuance of Amendment 279 (Reference 166). Since the requirements of this license condition have been completed, this license condition may be eliminated.</p>  |  |

| <b>License Condition 2.C(10)</b>  |   |
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| <u>Current License Condition 2.C(10)</u><br><br><u>Removed Details and Requirements Relocated to Other Controlled Documents</u> | <u>Proposed License Condition 2.C(10)</u><br><br><b><i>Deleted per Amendment No. ###.</i></b> |

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| <p>License Amendment No. 279 authorizes the relocation of certain technical specifications and operating license conditions, if applicable, to other licensee-controlled documents. Implementation of this amendment shall include relocation of these requirements to the specified documents.</p>   |  |
| <p style="text-align: center;"><b>Basis</b></p>   |  |
| <p>The removal of details and requirements relocated to other controlled documents as part of License Amendment 279 is proposed for deletion. The implementation of these requirements was completed in accordance with the schedule specified by the NRC on the issuance of Amendment 279 (Reference 166). Since the requirements of this license condition have been completed, this license condition may be eliminated.</p> |  |

| <p style="text-align: center;"><b>License Condition 2.C(11)</b></p>   |  |
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| <p><u>License Renewal License Conditions</u></p> <p>(a) The information in the Updated Final Safety Analysis Report (UFSAR) supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by the Commitments applicable to Davis-Besse Nuclear Power Station, Unit No. 1, in Appendix A of the "Supplemental Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station" (SER) dated August 2015, is collectively the "License Renewal UFSAR Supplement." The License Renewal 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 and activities applicable to Davis-Besse Nuclear Power Station, Unit No. 1, described in the License Renewal UFSAR Supplement provided the licensee evaluates such</p> | <p><u>Proposed License Condition 2.C(11)</u></p> <p><b><i>Deleted per Amendment No. ###.</i></b></p> |

changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(b) This License Renewal UFSAR Supplement, as revised per License Condition 11(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation.

1. The licensee shall implement those new programs and enhancements to existing programs no later than October 22, 2016.
2. The licensee shall complete those activities as noted in the Commitments applicable to Davis-Besse Nuclear Power Station, Unit No. 1, in the License Renewal UFSAR Supplement no later than October 22, 2016 or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.
3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

(c) This license condition requires testing of surveillance capsules for the period of extended operation to meet the test procedures and reporting requirements of American Society of Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. All pulled capsules shall be properly maintained for testing, and any changes to storage requirements must be

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| approved by the NRC. All pulled and tested capsules, unless discarded before August 31, 2000, shall be placed in storage to be saved for possible future reconstitution and use.   |  |
| <b>Basis</b>   |  |
| <p>License Condition 2.C.(11)(a) is a one-time requirement to update the UFSAR to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update as required by 10 CFR 50.71(e). Since the UFSAR update required by this license condition has been previously completed, this license condition has been satisfied and is therefore no longer needed. License Condition 2.C.(11)(a) also states that the licensee may make changes to the programs and activities described in the supplement without prior NRC approval provided that the changes are made pursuant to 10 CFR 50.59 requirements. The requirements of 10 CFR 50.59 will continue to apply to such changes after the license condition is deleted. Therefore, after deletion of this license condition, changes to these programs and activities may be made without prior NRC approval provided that FENOC evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59.</p> <p>License Condition 2.C.(11)(b) is a requirement to implement certain programs and activities prior to the period of extended operation. FENOC notified the NRC of the completion of this license condition in a letter dated November 18, 2016 (Reference 9). This license condition has been completed in its entirety and therefore is proposed for deletion.</p> <p>License Condition 2.C.(11)(c) is a license renewal requirement to test surveillance capsules for the period of extended operation, as well as the storage requirements for pulled and tested capsules. This license condition is no longer required as the reactor will be permanently defueled and will no longer be operated. It is, therefore, proposed for deletion.</p> |  |

| <b>License Condition 2.E.</b>   |  |
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| <u>Current License Condition 2.E.</u><br><br>This license is subject to the following antitrust conditions: <...> | <u>Proposed License Condition 2.E.</u><br><br><i><b>Deleted per Amendment No. ###.</b></i> |
| <b>Basis</b>  |  |

This license condition establishes the requirements for marketing or brokering of power or energy from DBNPS (antitrust conditions). This license condition and its related footnotes are proposed for deletion. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Since DBNPS will not be allowed to produce power, marketing or brokerage of power or energy becomes unfeasible. Therefore, it is acceptable to delete this license condition.

| License Condition 2.F.(1)   |   |
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| <u>Current License Condition 2.F.(1)</u>  | <u>Proposed License Condition 2.F.(1)</u>   |
| FENOC shall operate Davis-Besse Unit No. 1 within applicable Federal and State air and water quality standards.   | FENOC shall <del>operate</del> <b>-maintain</b> Davis-Besse Unit No. 1 within applicable Federal and State air and water quality standards. |
| Basis   |   |
| This license condition is changed to reflect the permanently defueled condition. The word “operate” will be replaced by the word “maintain.” Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant. |   |

| License Condition 2.F.(2)   |  |
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| <u>Current License Condition 2.F.(2)</u>  | <u>Proposed License Condition 2.F.(2)</u>  |
| Before engaging in an operational activity not evaluated by the Commission, FENOC will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement, FENOC shall provide a written evaluation of such | Before engaging in an <del>operational</del> activity not evaluated by the Commission, FENOC will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement, FENOC shall provide a written evaluation of such |



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| activities and obtain prior approval of the Director, Office of Nuclear Reactor Regulation for the activities.  | activities and obtain prior approval of the Director, Office of Nuclear Reactor Regulation for the activities. |
| <b>Basis</b>  |  |
| The word “operational” in this license condition is proposed for deletion. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled plant. |  |

| <b>License Condition 2.G.</b>   |  |
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| <u>Current License Condition 2.G.</u><br><br>In accordance with the requirement imposed by the October 8, 1976, order of the United States Court of Appeals for the District of Columbia Circuit in <u>Natural Resources Defense Council v. Nuclear Regulatory Commission</u> , No. 74-1385 and 74-1586, that the Nuclear Regulatory Commission “shall make any licenses granted between July 21, 1976 and such time when the mandate is issued subject to the outcome of such proceedings herein,” this license shall be subject to the outcome of such proceedings. | <u>Proposed License Condition 2.G.</u><br><br><i><b>Deleted per Amendment No. ###.</b></i> |
| <b>Basis</b>  |  |
| This may be deleted as the issues in this case were resolved generically by 10 CFR 51.51.   |  |

| License Condition 2.H.   |  |
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| <u>Current License Condition 2.H.</u><br><br>This renewed license is effective as of the date of issuance and shall expire at midnight April 22, 2037.   | <u>Proposed License Condition 2.H.</u><br><br>This <del>renewed</del> license is effective as of the date of issuance and <del>shall expire at midnight April 22, 2037</del> <b>is effective until the Commission notifies the licensee in writing that the license is terminated.</b> |
| Basis  |  |
| This license condition is revised to conform to 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession of the facility until the Commission notifies the licensee in writing that the license is terminated. |  |

| Proposed License Condition 2.I.  |   |
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| <u>Current License Condition 2.I.</u><br><br>[None]  | <u>Proposed License Condition 2.I.</u><br><br><b><i>Handling of irradiated fuel that has occupied part of a critical reactor core within the previous 95 days is not permitted.</i></b> |
| Basis  |   |
| <p>Once the reactor has been permanently defueled with all spent fuel placed in the SFP and the certifications submitted and docketed in accordance with 10 CFR 50.82, power operation or emplacement of fuel in the reactor will not be allowed. Therefore, all DBAs associated with power operations or fuel handling inside containment will no longer be applicable, which provides the basis for removal of the Safety Limits and most of the LCOs. The deletion of TS 3.3.14, Fuel Handling Exhaust – High Radiation, TS 3.7.10, Control Room Emergency Ventilation System (CREVS), and TS 3.7.13, Spent Fuel Pool Area Emergency Ventilation System (EVS), are based on the new FHA analysis, "Radiological Consequences of a Fuel Handling Accident Outside Containment After Permanent Shutdown," which was previously described in this submittal. This analysis removes credit for any of the requirements in the LCOs mentioned above during fuel handling activities. However, the analysis assumes the irradiated fuel has decayed for at least 95 days after reactor shutdown.</p> <p>In order to implement the PDTS prior to the 95-day decay time assumed in the new FHA analysis, FENOC proposes to prohibit movement of irradiated fuel that has occupied</p> |   |

part of a critical reactor core within the previous 95 days after permanently shutdown through the imposition of the proposed License Condition.

| License Condition 3.   |   |
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| <p><u>Current License Condition 3.</u></p> <p>Based on the Commission's Order dated December 16, 2005 and conforming Amendment No. 270 dated December 16, 2005 regarding the direct transfer of the license from the Cleveland Electric Illuminating Company (Cleveland Electric) and the Toledo Edison Company (Toledo Edison) to FirstEnergy Nuclear Generation Corp. (FENGenCo)*, FirstEnergy Nuclear Operating Company and FENGenCo* shall comply with the following conditions noted below:</p> | <p><u>Proposed License Condition 3.</u></p> <p>Based on the Commission's Order dated December 16, 2005 and conforming Amendment No. 270 dated December 16, 2005, <del>regarding the direct transfer of the license from the Cleveland Electric Illuminating Company (Cleveland Electric) and the Toledo Edison Company (Toledo Edison) to FirstEnergy Nuclear Generation Corp. (FENGenCo)*, FirstEnergy Nuclear Operating Company and FENGenCo*</del> <b>FirstEnergy Nuclear Generation LLC</b> shall comply with the following conditions noted below:</p> |
| Basis  |   |
| <p>The basis for these changes are that the transfer has been completed, and the name change has been incorporated from the footnote on license pages L-18 and L-19.</p>   |   |

| License Condition 3.A.   |  |
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| <p><u>Current License Condition 3.A.</u></p> <p>On the closing date of the transfers to FENGenCo* of their interests in Davis-Besse, Cleveland Electric and Toledo Edison shall transfer to FENGenCo* all of each transferor's respective accumulated decommissioning funds for Davis-Besse and tender to FENGenCo* additional amounts equal to remaining funds expected to be collected in 2005, as represented in the application dated June 1, 2005, but not yet collected by the</p> | <p><u>Proposed License Condition 3.A.</u></p> <p><del>On the closing date of the transfers to FENGenCo* of their interests in Davis-Besse, Cleveland Electric and Toledo Edison shall transfer to FENGenCo* all of each transferor's respective accumulated decommissioning funds for Davis-Besse and tender to FENGenCo* additional amounts equal to remaining funds expected to be collected in 2005, as represented in the application dated June 1, 2005, but not yet collected by the</del></p> |

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| <p>time of closing. All of the funds shall be deposited in a separate external trust fund for the reactor in the same amount as received with respect to the unit to be segregated from other assets of FENGenCo* and outside its administrative control, as required by NRC regulations, and FENGenCo* shall take all necessary steps to ensure that this external trust fund is maintained in accordance with the requirements of the order approving the transfer of the license and consistent with the safety evaluation supporting the order and in accordance with the requirements of 10 CFR Section 50.75, "Reporting and recordkeeping for decommissioning planning."</p> | <p><del>time of closing. All of the funds shall be deposited in a separate external trust fund for the reactor in the same amount as received with respect to the unit to be segregated from other assets of FENGenCo* and outside its administrative control, as required by NRC regulations, and FENGenCo*</del> <b>FirstEnergy Nuclear Generation, LLC</b> shall take all necessary steps to ensure that <del>this external the</del> <b>decommissioning</b> trust fund is maintained in accordance with the requirements of the <del>Order</del> approving the transfer of the license <b>dated December 16, 2005</b> and consistent with the safety evaluation supporting the <del>Order</del> and in accordance with the requirements of 10 CFR Section 50.75, "Reporting and recordkeeping for decommissioning planning."</p> |
| <p style="text-align: center;"><b>Basis</b></p>   |  |
| <p>The deletions are actions that have been completed and may therefore be removed from the license. As noted in the footnotes on license pages L-18 and L-19, FirstEnergy Nuclear Generation Corp. (FENGenCo)* has been renamed FirstEnergy Nuclear Generation, LLC.</p>   |  |

| <p style="text-align: center;"><b>License Condition 3.B.</b></p>   |   |
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| <p><u>Current License Condition 3.B.</u></p> <p>The Support Agreement described in the application dated June 1, 2005 (up to \$400 million), shall be effective consistent with the representations contained in the application. FENGenCo* shall take no action to cause FirstEnergy, or its successors and assigns, to void, cancel, or modify the Support Agreement without the prior written consent of the NRC staff. FENGenCo* shall inform the Director of the Office of Nuclear Reactor Regulation, in</p> | <p><u>Proposed License Condition 3.B.</u></p> <p><b><del>Deleted per Amendment No. ###.</del></b></p> |

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| writing, no later than ten days after any funds are provided to FENGenCo* by FirstEnergy under either Support Agreement.   |  |
| <b>Basis</b>   |  |
| <p>Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The Support Agreement will no longer be necessary to satisfy NUREG-1577 and 10 CFR 50.33(f) for operating nuclear power plants.</p> <p>As discussed in Section 1 of this LAR, the NRC is currently reviewing a licensing action for this condition. As License Condition 3.B is proposed for deletion in its entirety with this request, the licensing actions are unrelated.</p> |  |

| <b>Attachment 2 to License NPF-3</b>  |                              |
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| <u>Current Attachment 2</u>   | <u>Proposed Attachment 2</u> |
| Preoperational Tests, Startup Tests and Other Items Which Must Be Completed Prior to Proceeding to Succeeding Operational Modes   | <b><i>Deleted.</i></b>       |
| <b>Basis</b>  |                              |
| <p>Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Preoperational tests will no longer be required.</p> |                              |

| <b>TS Section 1.1 – Definitions</b>  |
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| <p>This section provides defined terms that are applicable throughout the TS and TS Bases. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). A number of the definitions are proposed for deletion because they have no relevance to and no longer apply to the permanently defueled facility status.</p> |

| Definitions Proposed for Deletion |  |
|-----------------------------------|--|
| <u>Term</u>                       | <u>Definition</u>  |
| ALLOWABLE THERMAL POWER           | ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.   |
| AXIAL POWER IMBALANCE             | AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP.  |
| AXIAL POWER SHAPING RODS (APSRs)  | APSRs shall be control components used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.   |
| CHANNEL CALIBRATION               | A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps. |
| CHANNEL CHECK                     | A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel   |



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|                                     | indication and status to other indications or status derived from independent instrument channels measuring the same parameter.  |
| CHANNEL FUNCTIONAL TEST             | A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps.  |
| CONTROL RODS                        | CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations.   |
| CORE OPERATING LIMITS REPORT (COLR) | The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.  |
| DOSE EQUIVALENT I-131               | DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or those listed in ICRP 30, Supplement to Part 1, page 192-212, table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity". |

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| Ē - AVERAGE DISINTEGRATION ENERGY | Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.  |
| INSERVICE TESTING PROGRAM         | The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).  |
| LEAKAGE                           | <p>LEAKAGE shall be:</p> <p>a. <u>Identified LEAKAGE</u></p> <ol style="list-style-type: none"> <li>1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal return flow), that is captured and conducted to collection systems or a sump or collecting tank;</li> <li>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</li> <li>3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE),</li> </ol> <p>b. <u>Unidentified LEAKAGE</u><br/>All LEAKAGE (except RCP seal return flow) that is not identified LEAKAGE; and</p> <p>c. <u>Pressure Boundary LEAKAGE</u><br/>LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.</p> |
| MODE<br>(Including Table 1.1-1)   | A MODE shall correspond to any one inclusive combination of core reactivity  |

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|   | condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.   |
| NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ( $F_Q$ )                | $F_Q$ shall be the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions.   |
| NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ( $F_{\Delta H}^N$ ) | $F_{\Delta H}^N$ shall be the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power.   |
| 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.<br>These tests are: <ul style="list-style-type: none"> <li>a. Described in Section 14, "Initial Tests and Operation," of the UFSAR;</li> <li>b. Authorized under the provisions of 10 CFR 50.59; or</li> <li>c. Otherwise approved by the Nuclear Regulatory Commission</li> </ul>  |
| PRESSURE AND TEMPERATURE LIMITS REPORTS (PTLR)                | The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current  |

|   |  |
|---|--|
|   | reactor vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.   |
| QUADRANT POWER TILT (QPT)                             | QPT shall be defined by the following equation and is expressed as a percentage of the Power in any Core Quadrant ( $P_{quad}$ ) to the Average Power of all Quadrants ( $P_{avg}$ ).<br>$QPT = 100 [ (P_{quad} / P_{avg}) - 1 ]$  |
| RATED THERMAL POWER (RTP)                             | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2817 MWt.   |
| REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME         | The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.  |
| SAFETY FEATURES ACTUATION SYSTEM (SFAS) RESPONSE TIME | The SFAS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its SFAS actuation setpoint at the channel sensor until the SFAS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. |
| SHUTDOWN MARGIN                                       | SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: <ul style="list-style-type: none"> <li>a. All full length CONTROL RODS (safety and regulating) are fully</li> </ul>   |

|  |  |
|--|--|
|  | <p>inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;</p> <ul style="list-style-type: none"> <li>b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and</li> <li>c. There is no change in APSR position.</li> </ul>             |
| STAGGERED TEST BASIS   | <p>A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, trains, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, trains, channels, or other designated components are tested during <math>n</math> Surveillance Frequency intervals, where <math>n</math> is the total number of systems, subsystems, trains, channels, or other designated components in the associated function.</p> |
| STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM (SFRCS) RESPONSE TIME | <p>The SFRCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its SFRCS actuation setpoint at the channel sensor until the SFRCS equipment is capable of performing its safety function (i.e., valves travel to their required positions, pumps discharge pressures reach their required values, etc.). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.</p>               |
| THERMAL POWER  | <p>THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.</p>  |

| <b>Basis</b>  |
|---|
| <p>These definitions are for terms that are relevant to power operation. Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, these terms will no longer be relevant to a facility in a permanently shutdown and defueled condition, and are not used in any PDTs specifications. Therefore, they are proposed for deletion.</p> |

| <b>TS Section 1.3 – Completion Times</b> |   |
|--|---|
| <u>Subsection</u>                        | <u>Description of Proposed Change</u>   |
| BACKGROUND                               | <p>This is proposed for revision to remove reference to “operation of the unit” and replace it with reference to “handling and storage of spent nuclear fuel.” Proposed changes are shown in Attachment 1.</p>  |
| DESCRIPTION                              | <p>This explanation is proposed for revision to remove discussion of MODES, which will not exist in a permanently defueled facility. Discussion is also removed for the senior licensed operator determining when the completion time begins. Discussion is also removed for entries into more than one condition, or alternating between conditions, as each of the three remaining permanently defueled specifications only has one condition. The term "unit" is typically associated with an operating reactor and is revised with the term "facility." This administrative change more appropriately represents the permanently shutdown and defueled condition. Proposed changes are shown in Attachment 1.</p> |
| EXAMPLES                                 | <p>The examples in this section are proposed for deletion. The examples are no longer necessary because they describe examples of Completion Times that do not remain in the PDTs. The Action that remains in the PDTs must be completed "Immediately," which is retained in PDTs Section 1.3.</p>  |



| Basis  |
|--|
| <p>Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2); therefore, certain terms currently provided in the TS no longer apply. Thus, TS Section 1.3 is being revised to be consistent with the permanently defueled condition.</p> |

| TS Section 1.4 – Frequency  |  |
|---|--|
| <u>Subsection</u>   | <u>Description of Proposed Change</u>  |
| DESCRIPTION   | <p>This is proposed for revision to remove discussion of surveillance performance situations that do not exist in the PDTs. Proposed changes are shown in Attachment 1.</p>  |
| EXAMPLES  | <p>This section is proposed for revision to remove discussion of surveillance performance situations that do not exist in the PDTs and to explicitly address those that do exist. An administrative change in this section replaces the term “unit” with the term “facility.” Reference to the term “MODE” is either deleted or replaced with terms such as “specified condition.” This former term (“MODE”) is no longer applicable to a permanently defueled facility. Examples 1.4-2 through 1.4-6 are proposed for deletion because these examples are not needed in a permanently defueled condition. Proposed changes are shown in Attachment 1.</p> |
| Basis   |  |
| <p>Once FENOC docket the certifications required by 10 CFR 50.82(a)(1), the 10 CFR 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, certain terms and examples currently provided in the TS no longer apply. Thus, TS Section 1.4 is being revised to be consistent with the permanently defueled condition.</p> |  |

| <b>TS Section 2.0 Safety Limits (SLs)</b>  |                          |
|--|--------------------------|
| <u>Current DBNPS TS</u>  | <u>Proposed DBNPS TS</u> |
| TS 2.1 – Safety Limits   | TS 2.1 – Deleted         |
| TS 2.2 – Safety Limit Violations   | TS 2.2 – Deleted         |
| <b>Basis</b>   |                          |
| <p>TS Section 2.1, Safety Limits, contains limits upon important process variables 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 reactor coolant system. The safety limits in this TS apply only to the reactor core and the reactor coolant system pressure. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Since the safety limits apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, the safety limits listed in TS Section 2.1, are no longer applicable.</p> <p>TS Section 2.2, Safety Limit Violations, directs actions to be taken if a safety limit specified in TS 2.1 is violated. TS 2.2 is applicable commensurate with the applicable MODES of the respective safety limits specified in TS 2.1. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Since the safety limits apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, the safety limit violations listed in TS Section 2.2 are no longer applicable.</p> <p><u>Summary:</u><br/>This section is proposed for deletion in its entirety, since the safety limits do not apply to a reactor that is in a permanently defueled condition. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which address the Safety Limits will no longer be applicable. Based on the above, the proposed deletion of all TS in Section 2.0 is acceptable, and the deletion of these TS will have no impact on continued safe maintenance of the facility. The corresponding TS bases will also be deleted.</p> |                          |

| <b>TS Section 3.0 Surveillance Requirements (SR) Applicability</b>   |                    |
|--|--------------------|
| Current DBNPS TS   | Proposed DBNPS TS  |
| SR 3.0.1   | SR 3.0.1 – Revised |
| SR 3.0.2   | SR 3.0.2 – Revised |
| SR 3.0.4   | SR 3.0.4 – Revised |
| <b>Basis</b>   |                    |
| <p>TS Section 3.0, SR Applicability, establishes the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.</p> <p>SR 3.0.1 is proposed for revision to delete references to the term “MODES.” Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, references to operating modes will no longer be relevant and therefore the term “MODES” is proposed for deletion.</p> <p>SR 3.0.2 is proposed for revision to remove conditions for periodic performance frequencies that do not exist in PDTs LCOs.</p> <p>SR 3.0.4 is proposed for revision to delete references to the term “MODES,” to delete the discussion pertaining to LCO 3.0.4, and to delete discussion pertaining to shutdown of the unit. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, references to operating modes and the discussion pertaining to shutdown of the unit will no longer be relevant and therefore are proposed for deletion. The discussion pertaining to LCO 3.0.4 is proposed for deletion, since LCO 3.0.4 is proposed for deletion within this document.</p> <p><u>Summary:</u><br/>Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the proposed revisions of SR</p> |                    |

3.0.1, 3.0.2, and 3.0.4 are acceptable and will not impact continued safe maintenance of the facility. The corresponding TS bases will also be revised accordingly.

| <b>TS Section 3.1 Reactivity Control Systems</b>  |                    |
|---|--------------------|
| Current DBNPS TS  | Proposed DBNPS TS  |
| TS 3.1.1 – SHUTDOWN MARGIN (SDM)  | TS 3.1.1 – Deleted |
| TS 3.1.2 – Reactivity Balance   | TS 3.1.2 – Deleted |
| TS 3.1.3 – Moderator Temperature Coefficient (MTC)  | TS 3.1.3 – Deleted |
| TS 3.1.4 – CONTROL ROD Group Alignment Limits   | TS 3.1.4 – Deleted |
| TS 3.1.5 – Safety Rod Insertion Limits  | TS 3.1.5 – Deleted |
| TS 3.1.6 – AXIAL POWER SHAPING ROD (APSR) Alignment Limits  | TS 3.1.6 – Deleted |
| TS 3.1.7 – Position Indicator Channels  | TS 3.1.7 – Deleted |
| TS 3.1.8 – PHYSICS TESTS Exceptions - MODE 1  | TS 3.1.8 – Deleted |
| TS 3.1.9 – PHYSICS TESTS Exceptions - MODE 2  | TS 3.1.9 – Deleted |
| <b>Basis</b>  |                    |
| <p>TS Section 3.1, Reactivity Control Systems, contains LCOs that provide for appropriate control of process variables, design features, or operating restrictions required to protect the integrity of a fission product barrier.</p> <p>The TS listed below do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS Bases are also proposed for deletion to reflect this change.</p> <p>TS 3.1.1, SHUTDOWN MARGIN (SDM), defines the minimum shutdown margin in the reactor core. The SDM limits are specified in the COLR. TS 3.1.1 is applicable in MODES 3, 4, and 5.</p> |                    |

TS 3.1.2, Reactivity Balance, defines the required accuracy for measured versus predicted core reactivity balance. TS 3.1.2 is applicable in MODES 1 and 2.

TS 3.1.3, Moderator Temperature Coefficient (MTC), identifies that the MTC shall be within specified limits in the COLR to ensure the core operates within the assumptions of the accident analysis. The MTC limits are specified in the COLR. The MTC limits specified in the COLR ensure that accidents that result in core overheating and overcooling will not violate the accident analysis assumptions. TS 3.1.3 is applicable in MODES 1 and 2.

TS 3.1.4, CONTROL ROD Group Alignment Limits, specifies requirements for limits on shutdown or control rod alignments, to ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The rod OPERABILITY requirements (trippability) are separate from the alignment requirements. The rod OPERABILITY requirement is satisfied provided the rod will insert in the required rod drop time assumed in the safety analysis. TS 3.1.4 is applicable in MODES 1 and 2.

TS 3.1.5, Safety Rod Insertion Limits, specifies that safety rods must be fully withdrawn any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. TS 3.1.5 is applicable in MODES 1 and 2.

TS 3.1.6, AXIAL POWER SHAPING ROD (APSR) Alignment Limits, specifies the limits for axial power shaping control rod alignments. TS 3.1.6 is applicable in MODES 1 and 2.

TS 3.1.7, Position Indicator Channels, defines the operability requirements for control rod absolute position indicator and relative position indicator channels for control rods and APSRs. This ensures that CONTROL ROD and APSR position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. TS 3.1.7 is applicable in MODES 1 and 2.

TS 3.1.8, PHYSICS TESTS Exceptions - MODE 1, specifies conditions to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. TS 3.1.8 is applicable in MODE 1 during PHYSICS TESTS.

TS 3.1.9, PHYSICS TESTS Exceptions - MODE 2, specifies conditions to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. TS 3.1.9 is applicable in MODE 2 during PHYSICS TESTS.

Summary:

The above TSs are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 5. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which only address their associated specific plant equipment, control of process variables, design features, or operating restrictions are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.1 is acceptable with no impact on continued safe maintenance of the facility. With the TS section deleted in its entirety, the corresponding TS bases will also be deleted accordingly.

| <b>TS Section 3.2 Power Distribution Limits</b>  |                    |
|--|--------------------|
| Current DBNPS TS   | Proposed DBNPS TS  |
| TS 3.2.1 – Regulating Rod Insertion Limits   | TS 3.2.1 – Deleted |
| TS 3.2.2 – AXIAL POWER SHAPING ROD (APSR) Insertion Limits   | TS 3.2.2 – Deleted |
| TS 3.2.3 – AXIAL POWER IMBALANCE Operating Limits  | TS 3.2.3 – Deleted |
| TS 3.2.4 – QUADRANT POWER TILT (QPT)   | TS 3.2.4 – Deleted |
| TS 3.2.5 – Power Peaking Factors   | TS 3.2.5 – Deleted |
| <b>Basis</b>   |                    |
| <p>TS Section 3.2, Power Distribution Limits, contains LCOs that provide for appropriate control of process variables, design features, or operating restrictions required to protect the integrity of a fission product barrier.</p> <p>The TS listed below do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS Bases are also proposed for deletion to reflect this change.</p> <p>TS 3.2.1, Regulating Rod Insertion Limits, specifies the insertion, sequence, and overlap limits for regulating control rods. The limits on regulating rod sequence, including group overlap, and insertion positions must be maintained because they ensure that the resulting power distribution is within the range of analyzed power distributions and that the SDM and ejected rod worth are maintained. These limits are specified in the COLR. TS 3.2.1 is applicable in MODES 1 and 2.</p> |                    |



TS 3.2.2, AXIAL POWER SHAPING ROD (APSR) Insertion Limits, identifies that the APSRs will be positioned according to the limits in the COLR. The limits on APSR physical insertion, as defined in the COLR, must be maintained because they serve the function of controlling the power distribution within an acceptable range. TS 3.2.2 is applicable in MODES 1 and 2.

TS 3.2.3, AXIAL POWER IMBALANCE Operating Limits, identifies that Axial Power Imbalance in the core shall be maintained within the acceptable operating limits specified in the COLR. This LCO is required to limit the core power distribution based on accident initial condition criteria. The power density at any point in the core must be limited to maintain specified acceptable fuel design limits. This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the  $F_Q$  and  $F_{\Delta H}^N$  limits given in the COLR. TS 3.2.3 is applicable in MODE 1 with THERMAL POWER greater than 40% RTP.

TS 3.2.4, QUADRANT POWER TILT (QPT), identifies that QPT shall be maintained less than or equal to the steady state limits specified in the COLR. This LCO is required to limit the core power distribution based on accident initial condition criteria. The power density at any point in the core must be limited to maintain specified acceptable fuel design limits. Together, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 provide limits on control component operation and on monitored process variables to ensure that the core operates within the  $F_Q$  and  $F_{\Delta H}^N$  limits given in the COLR. TS 3.2.4 is applicable in MODE 1 with THERMAL POWER greater than 20% RTP.

TS 3.2.5, Power Peaking Factors, identifies that  $F_Q$  and  $F_{\Delta H}^N$  shall be within the limits specified in the COLR. This LCO ensures that the core operates within the bounds assumed for the emergency core cooling systems (ECCS) and thermal hydraulic analyses. TS 3.2.5 is applicable in MODE 1 with THERMAL POWER greater than 20% RTP.

Summary:

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 and 2. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which only address their associated specific plant equipment, control of process variables, design features, or operating restrictions are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.2 is acceptable with no impact on continued safe maintenance of the facility. With the TS section deleted in its entirety, the corresponding TS bases will also be deleted accordingly.

| <b>TS Section 3.3 Instrumentation</b>  |                     |
|--|---------------------|
| Current DBNPS TS   | Proposed DBNPS TS   |
| TS 3.3.1 – Reactor Protection System (RPS) Instrumentation                       | TS 3.3.1 – Deleted  |
| TS 3.3.2 – Reactor Protection System (RPS) Manual Reactor Trip                   | TS 3.3.2 – Deleted  |
| TS 3.3.3 – Reactor Protection System (RPS) - Reactor Trip Module (RTM)           | TS 3.3.3 – Deleted  |
| TS 3.3.4 – CONTROL ROD Drive (CRD) Trip Devices                                  | TS 3.3.4 – Deleted  |
| TS 3.3.5 – Safety Features Actuation System (SFAS) Instrumentation               | TS 3.3.5 – Deleted  |
| TS 3.3.6 – Safety Features Actuation System (SFAS) Manual Initiation             | TS 3.3.6 – Deleted  |
| TS 3.3.7– Safety Features Actuation System (SFAS) Automatic Actuation Logic      | TS 3.3.7 – Deleted  |
| TS 3.3.8 – Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)           | TS 3.3.8 – Deleted  |
| TS 3.3.9 – Source Range Neutron Flux   | TS 3.3.9 – Deleted  |
| TS 3.3.10 – Intermediate Range Neutron Flux                                      | TS 3.3.10 – Deleted |
| TS 3.3.11 – Steam and Feedwater Rupture Control System (SFRCS) Instrumentation   | TS 3.3.11 – Deleted |
| TS 3.3.12 – Steam and Feedwater Rupture Control System (SFRCS) Manual Initiation | TS 3.3.12 – Deleted |
| TS 3.3.13 – Steam and Feedwater Rupture Control System (SFRCS) Actuation         | TS 3.3.13 – Deleted |
| TS 3.3.14 – Fuel Handling Exhaust - High Radiation                               | TS 3.3.14 – Deleted |
| TS 3.3.15 – Station Vent Normal Range Radiation Monitoring                       | TS 3.3.15 – Deleted |
| TS 3.3.16 – Anticipatory Reactor Trip System (ARTS) Instrumentation              | TS 3.3.16 – Deleted |

|   |                     |
|---|---------------------|
| TS 3.3.17 – Post Accident Monitoring (PAM) Instrumentation  | TS 3.3.17 – Deleted |
| TS 3.3.18 – Remote Shutdown System  | TS 3.3.18 – Deleted |
| <b>Basis</b>  |                     |
| <p>TS Section 3.3, Instrumentation, contains LCOs that provide for appropriate functional capability of sensing and control instrumentation required for safe operation of the facility.</p> <p>The TS listed below do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS Bases are also proposed for deletion to reflect this change.</p> <p>TS 3.3.1, Reactor Protection System (RPS) Instrumentation, identifies the requirements for the operability of RPS channels for each RPS function as specified in the associated Table 3.3.1-1. The RPS initiates a reactor trip to protect against violating the core fuel design limits and the RCS pressure boundary during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the SFAS in mitigating accidents. The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the LCOs on other reactor system parameters and equipment performance. TS 3.3.1 is applicable in MODES 1, 2, 3, 4, and 5 (according to specific applicability requirements for each RPS function listed in TS Table 3.3.1-1).</p> <p>TS 3.3.2, Reactor Protection System (RPS) Manual Reactor Trip, identifies the conditions under which the RPS manual trip function shall be operable. The manual reactor trip ensures that the control room operator can initiate a reactor trip at any time. The manual reactor trip channels are required as a backup to the automatic trip functions and allows operators to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. TS 3.3.2 is applicable in MODES 1 and 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.</p> <p>TS 3.3.3, Reactor Protection System (RPS) - Reactor Trip Module (RTM), identifies the requirements for the operability of RPS RTMs. Accident analyses rely on a reactor trip for protection of reactor core integrity and reactor coolant pressure boundary integrity. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. Four RTMs must be OPERABLE to ensure that a reactor trip would occur if needed any time the reactor is critical. OPERABILITY is defined as the RTM being able to receive and interpret trip signals from its own and other RPS channels and to open its associated trip devices (CRD trip breaker or silicon controlled rectifier (SCR) relays, as applicable). The requirement of four channels to be OPERABLE ensures that no single RTM failure can preclude an RPS trip via the CRD</p> |                     |

trip breakers. TS 3.3.3 is applicable in MODES 1 and 2, and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

TS 3.3.4, CONTROL ROD Drive (CRD) Trip Devices, identifies the conditions under which the CRD trip devices shall be operable. Accident analyses rely on a reactor trip for protection of reactor core integrity and reactor coolant pressure boundary integrity. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The control rod insertion limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS will trip when required. The RPS contains two types of CRD trip devices: four CRD trip breakers and two SCR relay trip channels. The LCO requires all of the CRD trip devices to be OPERABLE. Requiring four CRD trip breakers to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure. Requiring two SCR relay trip channels to be OPERABLE provides an additional method to interrupt power in each pathway to the CRDs. Requiring all devices OPERABLE also ensures that a single failure will not cause an unwanted reactor trip. TS 3.3.4 is applicable in MODES 1 and 2 and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD system is capable of rod withdrawal.

TS 3.3.5, Safety Features Actuation System (SFAS) Instrumentation, identifies the requirements for operability for each of the parameters listed in Table 3.3.5-1. Accident analyses rely on automatic SFAS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include loss of coolant accident (LOCA) and main steam line break (MSLB) events that result in RCS inventory reduction. The LCO requires four channels of SFAS instrumentation for each parameter to be OPERABLE in each SFAS train. TS 3.3.5 is applicable in MODES 1, 2, 3, and 4 (according to specific applicability requirements for each SFAS function listed in TS Table 3.3.5-1).

TS 3.3.6, Safety Features Actuation System (SFAS) Manual Initiation, identifies the requirements for operability of manual initiation channels of SFAS Functions. The SFAS manual initiation ensures that the control room operator can rapidly initiate engineered safety features (ESF) Functions at any time. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate SFAS whenever any parameter is rapidly trending toward its trip setpoint. Two SFAS manual initiation channels of each SFAS Function shall be OPERABLE whenever conditions exist that could require ESF protection of the reactor or containment. TS 3.3.6 is applicable in MODES 1, 2, and 3, and in MODE 4 when the associated engineered safety features equipment is required to be OPERABLE.

TS 3.3.7, Safety Features Actuation System (SFAS) Automatic Actuation Logic, identifies the requirements for SFAS automatic actuation logic matrices to be

OPERABLE. Accident analyses rely on automatic SFAS actuation for protection of the core and containment and for limiting off site dose levels following an accident. These include LOCA and MSLB events that result in RCS inventory reduction. The automatic actuation logic is an integral part of the SFAS. The automatic actuation output logic for each component actuated by the SFAS is required to be OPERABLE whenever conditions exist that could require ESF protection of the reactor or the containment. TS 3.3.7 is applicable in MODES 1, 2, and 3, and in MODE 4 when associated engineered safety features equipment is required to be OPERABLE.

TS 3.3.9, Source Range Neutron Flux, identifies the requirements for source range neutron flux channels to be OPERABLE. The source range neutron flux channels are necessary to monitor core reactivity changes. It is the primary means for detecting and triggering operator actions to respond to reactivity transients initiated from conditions when the RPS is not required to be operable. It also triggers operator actions to anticipate RPS actuation in the event of reactivity transients during startup and shutdown conditions. Two source range neutron flux channels (the channels associated with the RPS) shall be OPERABLE whenever the control rods are capable of being withdrawn to provide the operator with redundant source range neutron instrumentation. TS 3.3.9 is applicable in MODES 2, 3, 4, and 5.

TS 3.3.10, Intermediate Range Neutron Flux, identifies the requirements for intermediate range neutron flux channels to be OPERABLE. Intermediate range neutron flux channels are necessary to monitor core reactivity changes and are the primary indication to trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from low power conditions. Two intermediate range neutron flux instrumentation channels shall be OPERABLE to provide the operator with redundant neutron flux indication. These enable operators to control the increase in power and to detect neutron flux transients. TS 3.3.10 is applicable in MODE 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD system capable of rod withdrawal.

TS 3.3.11, Steam and Feedwater Rupture Control System (SFRCS) Instrumentation, identifies the requirements for the SFRCS instrumentation channels to be OPERABLE in accordance with associated Table 3.3.11-1. The SFRCS is designed to automatically start the auxiliary feedwater (AFW) system in the event of a MSLB, main feedwater (MFW) line rupture, a low level in the steam generators or a loss of all four reactor coolant pumps. SFRCS is designed to automatically isolate the main steam system and MFW system in the event of a MSLB or MFW line rupture. The AFW system is automatically aligned to feed the unaffected steam generator (SG) upon a loss of steam pressure in one of the SGs. All instrumentation performing an SFRCS system Function in Table 3.3.11-1 shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. Four channels are required OPERABLE for all SFRCS instrumentation Functions as specified in Table 3.3.11-1 to ensure that no single failure prevents actuation of a train. TS 3.3.11

is applicable in MODES 1, 2, and 3 (according to specific applicability requirements for each RPS function listed in TS Table 3.3.11-1).

TS 3.3.12, Steam and Feedwater Rupture Control System (SFRCS) Manual Initiation, identifies the requirements for the manual initiation switches to be OPERABLE for each SFRCS Function (auxiliary feedwater pump turbine 1 initiation, auxiliary feedwater pump turbine 2 initiation, auxiliary feedwater pump turbine 1 initiation and steam generator 1 isolation; and auxiliary feedwater pump turbine 2 initiation and steam generator 2 isolation). The SFRCS manual initiation capability provides the operator with the capability to actuate SFRCS Functions from the control room in the absence of any other initiation condition. SFRCS Functions credited in the safety analysis are automatic. However, the manual initiation Functions are required by design as backups to the automatic trip Functions and allow operators to initiate auxiliary feedwater pump turbine (AFPT) and actuate steam generator (SG) isolation whenever these Functions are needed. Each push button performing an SFRCS manual initiation Function shall be OPERABLE. Failure of any push button renders the affected Function inoperable. TS 3.3.12 is applicable in MODES 1, 2, and 3.

TS 3.3.13, Steam and Feedwater Rupture Control System (SFRCS) Actuation, identifies the requirements for the automatic actuation logic channels to be OPERABLE for each SFRCS Function (auxiliary feedwater initiation, auxiliary feedwater and main steam valve control, main steam line isolation, and main feedwater isolation). These SFRCS Functions are credited in the event of a MSLB or a MFW line break. TS 3.3.13 is applicable in MODES 1, 2, and 3.

TS 3.3.16, Anticipatory Reactor Trip System (ARTS) Instrumentation, identifies the requirements for the operability of ARTS channels for each ARTS Function as specified in the associated Table 3.3.16-1. The ARTS instrumentation initiates a reactor trip when a sensed parameter exceeds its setpoint value, indicating the approach of an unsafe condition thereby reducing the magnitude of pressure and temperature transients on the RCS caused by loss of main feedwater events or turbine trips. Four separate redundant protection channels receive inputs of MFW pump status and turbine status. TS 3.3.16 is applicable in MODE 1 (according to specific applicability requirements for each ARTS function listed in TS Table 3.3.16-1).

TS 3.3.17, Post Accident Monitoring (PAM) Instrumentation, identifies the PAM instrumentation that must be post accident monitoring operable as shown in Table 3.3.17-1. The PAM instrumentation displays unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis events. These essential instruments address the recommendations of Regulatory Guide 1.97 as required by Supplement 1 to NUREG-0737. The OPERABILITY of the PAM instrumentation ensures there is



sufficient information available on selected unit parameters to monitor and assess unit status following an accident. TS 3.3.17 is applicable in MODES 1, 2, and 3.

TS 3.3.18, Remote Shutdown System, identifies the OPERABILITY requirements of remote shutdown monitoring instrumentation, control circuits and transfer switch functions to place and maintain the unit in MODE 3. The remote shutdown monitoring instrumentation provides the control room operator with sufficient instrumentation to support maintaining the unit in a safe shutdown condition from locations other than the control room. TS 3.3.18 is applicable in Modes 1, 2, and 3.

TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.7, TS 3.3.9, TS 3.3.10, TS 3.3.11, TS 3.3.12, TS 3.3.13, TS 3.3.16, TS 3.3.17 and TS 3.3.18 are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 5. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.7, TS 3.3.9, TS 3.3.10, TS 3.3.11, TS 3.3.12, TS 3.3.13, TS 3.3.16, TS 3.3.17, and TS 3.3.18, which only address these specific plant systems, control of process variables, design features, or operating restrictions will no longer be applicable and may be deleted with no impact on continued safe maintenance of the facility. The corresponding TS bases for TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.7, TS 3.3.8, TS 3.3.9, TS 3.3.10, TS 3.3.11, TS 3.3.12, TS 3.3.13, TS 3.3.16, TS 3.3.17, and TS 3.3.18 are also being deleted to reflect this change.

The following TS from TS Section 3.3 are also being proposed for deletion as follows:

TS 3.3.8, Emergency Diesel Generator (EDG) Loss of Power Start (LOPS), identifies the conditions under which the loss of voltage and degraded voltage function channels are required to be OPERABLE. The EDG LOPS is required for the ESF to function in any accident with a loss of offsite power. Its design basis is that of the SFAS. The required channels of LOPS, in conjunction with the ESF systems powered from the EDGs, provide unit protection in the event of any of the analyzed accidents discussed in the UFSAR in which a loss of offsite power is assumed. The response times for SFAS actuated equipment in LCO 3.3.5, "Safety Features Actuation System (SFAS) Instrumentation," include the appropriate EDG loading and sequencing delay. TS 3.3.8 is applicable in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation is also required whenever the EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown," so that the EDG can perform its function on a loss of power or degraded power to the essential bus.



TS 3.3.8 provides SFAS Functions that are only applicable in MODES 1, 2, 3, and 4. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2).

TS 3.3.8 is also applicable to LCO 3.8.2 to ensure that the EDG can perform its function during a loss of power. However, as discussed in this document, TS 3.8.2 is proposed for deletion based on the EDG no longer being required to mitigate DBAs. Without a need for the EDG to remain operable, TS 3.3.8 is no longer needed. Therefore, TS 3.3.8 will no longer be applicable once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii). TS 3.3.8 may be deleted with no impact on continued safe maintenance of the facility. The corresponding TS basis for TS 3.3.8 is also being deleted to reflect this change.

TS 3.3.14, Fuel Handling Exhaust – High Radiation, identifies the requirements for the fuel handling exhaust – high radiation channels to be OPERABLE. The spent fuel pool area emergency ventilation system (EVS) actuation aligns the ventilation flow path through the high efficiency particulate air (HEPA) and charcoal filters prior to discharging to the station vent. TS 3.3.14 is applicable during movement of irradiated fuel assemblies in the spent fuel pool building. The fuel handling exhaust - high radiation Function has been assumed in the fuel handling accident outside containment analysis. Two fuel handling exhaust - high radiation channels are required to be OPERABLE during movement of irradiated fuel assemblies in the spent fuel pool building to ensure radiation doses are within the limits of the accident analyses. Filtration of the exhaust ensures the accident dose at the site boundary will be well below the 10 CFR 100 limits.

TS 3.3.15, Station Vent Normal Range Radiation Monitoring, identifies the requirements for the station vent normal range radiation monitoring instrumentation to be OPERABLE. The principal function of the Station Vent Normal Range Radiation Monitoring instrumentation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. The high radiation isolation function provides assurance that under the required conditions, an isolation signal will be given. The radiation monitors located in the station vent stack provide isolation and shutdown of the control room normal ventilation system. The control room isolation capability on high radiation shall be OPERABLE whenever there is a chance for a significant accidental release of radioactivity. This includes MODES 1, 2, 3, and 4, and during movement of irradiated fuel. If a radioactive release were to occur during any of these conditions, the control room would have to remain habitable to ensure reactor shutdown and cooling can be controlled from the control room. The fuel handling accident analyses, both inside and outside of containment, assume the control room normal ventilation system is isolated by the station vent normal range radiation monitoring high radiation signal.

The HEPA and charcoal filters are not credited in the updated fuel handling accident analysis discussed in this amendment request. This updated fuel handling accident analysis demonstrates that accident doses remain below the acceptance criteria without crediting any active component. The FHA analysis results determined that at least 95 days of irradiated fuel decay time after reactor shutdown and compliance with the spent fuel pool water level requirements of TS 3.7.14 are required for the CR, EAB, and LPZ doses to remain below the acceptance criteria. Therefore, TS 3.3.14 is no longer applicable after this period of time and may be deleted with no impact on continued safe maintenance of the facility. The isolation of the control room normal ventilation system is not credited in the updated fuel handling accident. As described in this amendment request, FENOC will sample the contents of the waste gas decay tank prior to implementing the PDTS to ensure that the activity of the contents is less than those applied in the FHA analysis. Upon permanent shutdown and cooldown, the source term contained within the waste gas decay tank represents the highest (worst case) source term and is expected to be significantly less than that assumed in the WGDTR analysis. Subsequent additions to the waste gas decay tank resulting from water management activities would be less than the final shutdown and cooldown waste gas tank source term. Therefore, TS 3.3.15 is no longer applicable and may be deleted with no impact on continued safe maintenance of the facility. The corresponding TS bases for TS 3.3.14 and 3.3.15 are also being deleted to reflect this change.

Summary:

The above TS are related to assuring the appropriate functional capability of sensing and control instrumentation required for safe operation of the facility. These TS do not apply to the safe storage and handling of spent fuel in the SFP.

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which address the capability of sensing and control instrumentation required for the safe operation of the facility are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.3 is acceptable with no impact on continued safe maintenance of the facility. With the TS section deleted in its entirety, the corresponding TS bases will also be deleted accordingly.

| <b>TS Section 3.4 Reactor Coolant System (RCS)</b>  |                     |
|---|---------------------|
| Current DBNPS TS  | Proposed DBNPS TS   |
| TS 3.4.1 – RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits | TS 3.4.1 – Deleted  |
| TS 3.4.2 – RCS Minimum Temperature for Criticality  | TS 3.4.2 – Deleted  |
| TS 3.4.3 – RCS Pressure and Temperature (P/T) Limits  | TS 3.4.3 – Deleted  |
| TS 3.4.4 – RCS Loops – MODES 1 and 2  | TS 3.4.4 – Deleted  |
| TS 3.4.5 – RCS Loops – MODE 3   | TS 3.4.5 – Deleted  |
| TS 3.4.6 – RCS Loops – MODE 4   | TS 3.4.6 – Deleted  |
| TS 3.4.7 – RCS Loops – MODE 5, Loops Filled   | TS 3.4.7 – Deleted  |
| TS 3.4.8 – RCS Loops – MODE 5, Loops Not Filled   | TS 3.4.8 – Deleted  |
| TS 3.4.9 – Pressurizer  | TS 3.4.9 – Deleted  |
| TS 3.4.10 – Pressurizer Safety Valves   | TS 3.4.10 – Deleted |
| TS 3.4.11 – Pressurizer Pilot Operated Relief Valve (PORV)                                  | TS 3.4.11 – Deleted |
| TS 3.4.12 – Low Temperature Overpressure Protection (LTOP)                                  | TS 3.4.12 – Deleted |
| TS 3.4.13 – RCS Operational LEAKAGE   | TS 3.4.13 – Deleted |
| TS 3.4.14 – RCS Pressure Isolation Valve (PIV) Leakage                                      | TS 3.4.14 – Deleted |
| TS 3.4.15 – RCS Leakage Detection Instrumentation   | TS 3.4.15 – Deleted |
| TS 3.4.16 – RCS Specific Activity   | TS 3.4.16 – Deleted |
| TS 3.4.17 – Steam Generator (SG) Tube Integrity   | TS 3.4.17 – Deleted |

| Basis  |
|--|
| <p>TS Section 3.4, Reactor Coolant System (RCS), contains LCOs 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.</p> <p>The TS listed below do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS bases are also proposed for deletion to reflect this change.</p> <p>TS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, specifies process variables requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the analyzed transients. TS 3.4.1 is applicable in MODE 1.</p> <p>TS 3.4.2, RCS Minimum Temperature for Criticality, identifies that each RCS loop average temperature shall be greater than or equal to 525 degrees Fahrenheit (°F). This LCO is to prevent criticality much outside the minimum normal operating regime (532°F) and to prevent operation in an unanalyzed condition. The limit of 525°F has been selected to be within the instrument indicating range (520°F to 620°F). The limit is also set slightly below the lowest power range operating temperature (532°F). TS 3.4.2 is applicable in MODE 1 and MODE 2 with <math>k_{eff}</math> greater than or equal to 1.0.</p> <p>TS 3.4.4, RCS Loops - MODES 1 and 2, identifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODES 1 and 2. The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators, to the secondary plant. The intent of the specification is to require core heat removal with forced flow during power operation. TS 3.4.4 is applicable in MODES 1 and 2.</p> <p>TS 3.4.5, RCS Loops - MODE 3, identifies that two RCS loops shall be OPERABLE and one RCS loop shall be in operation to ensure heat removal capability of the RCS loops with the reactor in MODE 3. The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators, to the secondary plant fluid. TS 3.4.5 is applicable in MODE 3.</p> <p>TS 3.4.6, RCS Loops - MODE 4, identifies that two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and one loop shall be in operation to ensure heat removal capability of the RCS loops with the reactor in MODE 4. The primary function of the reactor coolant in MODE 4 is removal of decay heat and transfer of this heat to the steam generators or decay heat removal (DHR) heat exchangers. TS 3.4.6 is applicable in MODE 4.</p> |

TS 3.4.7, RCS Loops - MODE 5, Loops Filled, identifies that two loops consisting of any combination of RCS loops and DHR loops shall be OPERABLE and one loop shall be in operation to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops filled with coolant. In MODE 5 with RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator secondary side coolant or the component cooling water via the DHR coolers. While the principal means for decay heat removal is via the DHR system, the steam generators are specified as a backup means for redundancy. TS 3.4.7 is applicable in MODE 5 with RCS loops filled.

TS 3.4.8, RCS Loops - MODE 5, Loops Not Filled, identifies that two DHR loops shall be OPERABLE and one DHR loop shall be in operation to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops not filled with coolant. In MODE 5 with loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the DHR coolers. The steam generators are not available as a heat sink when the loops are not filled. TS 3.4.8 is applicable in MODE 5 with the RCS loops not filled.

TS 3.4.9, Pressurizer, identifies OPERABILITY requirements for the RCS pressurizer. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. In MODES 1 and 2, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. TS 3.4.9 is applicable in MODES 1, 2, and 3.

TS 3.4.10, Pressurizer Safety Valves, identifies OPERABILITY and lift setpoint parameters for the pressurizer safety valves. The pressurizer safety valves provide, in conjunction with the reactor protection system, overpressure protection for the RCS. Two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for portions of MODE 3. For the remainder of MODE 3, MODES 4 and 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)." TS 3.4.10 is applicable in MODES 1, 2, and 3.

TS 3.4.11, Pressurizer Pilot Operated Relief Valve (PORV), identifies that the PORV and associated block valve shall be OPERABLE. The PORV is an electromatic pilot operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually operated using controls installed in the control room. An electric motor operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. The block valve is to be used to isolate a stuck open PORV to isolate the resulting small break LOCA.

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. TS 3.4.11 is applicable in MODES 1, 2, and 3.

TS 3.4.12, Low Temperature Overpressure Protection (LTOP), specifies that the DHR system relief valve shall be OPERABLE with certain conditions. The LTOP controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity through the DHR system relief valve. The DHR system relief valve provides overpressure protection for the RCS during low temperature operations. RCS and DHR systems are monitored for temperature and pressure. Maintaining the relief setpoint within the limits of the LCO ensures the 10 CFR 50, Appendix G, limits will be met in any event in the LTOP analysis. TS 3.4.12 is applicable in MODES 4 and 5, and in MODE 6 when the reactor vessel head is on.

TS 3.4.13, RCS Operational LEAKAGE, identifies process variable limits and operating restrictions for RCS pressure boundary leakage, unidentified RCS leakage, identified RCS leakage, and primary to secondary leakage. RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a design basis event. Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from an MSLB accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid. TS 3.4.13 is applicable in MODES 1, 2, 3, and 4.

TS 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage, identifies process variable limits and operating restrictions for RCS PIV leakage and the DHR system interlock function. 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A, discuss reactor coolant pressure boundary valves, which are normally closed valves in series within the RCPB boundary that separate the high pressure RCS from an attached low pressure system. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. PIVs are provided to isolate the RCS from the DHR system. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. TS 3.4.14 is applicable in MODES 1, 2, and 3, and in MODE 4, except the valves in the DHR flow path when in, or during the transition to or from, the DHR mode of operation and the DHR system interlock function.

TS 3.4.15, RCS Leakage Detection Instrumentation, identifies the RCS leakage detection instruments that are required to be OPERABLE and in what conditions. Leakage detection systems must have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified leakage. This LCO requires instruments of



diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified leakage are detected in time to allow actions to place the plant in a safe condition when RCS leakage indicates possible RCPB degradation. TS 3.4.15 is applicable in MODES 1, 2, 3, and 4.

TS 3.4.16, RCS Specific Activity, identifies process variable limits for dose equivalent I-131 and gross specific activity. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a SGTR accident. TS 3.4.16 is applicable in MODES 1 and 2, and MODE 3 with RCS average temperature ( $T_{avg}$ ) greater than or equal to 530°F.

TS 3.4.17, Steam Generator (SG) Tube Integrity, identifies requirements to ensure the RCPB integrity function of the SG. The SGTR accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this specification. TS 3.4.17 is applicable in MODES 1, 2, 3, and 4.

The TS above do not apply with the reactor defueled and are being proposed for deletion. The above TS are related to assuring the appropriate functional capability of plant equipment, control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 6. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which address plant equipment associated with the reactor coolant system, will no longer be applicable. Based on the above, the proposed deletion of the TS in Section 3.4 is acceptable. The corresponding TS bases will also be deleted.

The following TS is also proposed for deletion:

TS 3.4.3, RCS Pressure and Temperature (P/T) Limits, identifies that the RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits as specified in the Pressure - Temperature Limits Report (PTLR). 10 CFR 50, Appendix G, requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. This LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the RCPB. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. TS 3.4.3 is applicable at all times.

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in



accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). As such, the requirements of 10 CFR 50, Appendix G, no longer apply in such a condition because the reactor coolant pressure boundary will no longer be used as a fission product barrier when the reactor vessel is permanently defueled. Therefore, TS 3.4.3 is no longer needed and may be deleted with no impact on continued safe maintenance of the facility. The corresponding TS bases will also be deleted.

Summary:

The above TS are related to assuring the 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. These TS do not apply to the safe storage and handling of spent fuel in the SFP.

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Based on the above, the proposed deletion of all TS in Section 3.4 is acceptable with no impact on continued safe maintenance of the facility. With the TS section deleted in its entirety, the corresponding TS bases will also be deleted accordingly.

| <b>TS Section 3.5 Emergency Core Cooling Systems (ECCS)</b>  |                    |
|--|--------------------|
| Current DBNPS TS   | Proposed DBNPS TS  |
| TS 3.5.1 – Core Flooding Tanks (CFTs)  | TS 3.5.1 – Deleted |
| TS 3.5.2 – ECCS - Operating  | TS 3.5.2 – Deleted |
| TS 3.5.3 – ECCS - Shutdown   | TS 3.5.3 – Deleted |
| TS 3.5.4 – Borated Water Storage Tank (BWST)   | TS 3.5.4 – Deleted |
| <b>Basis</b>   |                    |
| <p>TS Section 3.5, Emergency Core Cooling Systems (ECCS), contains LCOs that provide for appropriate functional capability of ECCS equipment required for safe operation of the facility.</p> <p>The TS listed below do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS bases are also proposed for deletion to reflect this change.</p> |                    |

TS 3.5.1, Core Flooding Tanks (CFTs), identifies that two CFTs shall remain operable. The CFTs supply water to the reactor vessel during the blowdown phase of a LOCA, provide inventory during the refill phase that follows thereafter, and provide Reactor Coolant System (RCS) makeup for a small break LOCA. TS 3.5.1 is applicable in MODES 1 and 2, and MODE 3 with RCS pressure greater than 800 psig.

TS 3.5.2, ECCS - Operating, identifies that two ECCS trains shall remain operable. The ECCS trains provide core cooling during injection and recirculation modes through the use of high pressure injection (HPI) and low pressure injection (LPI) subsystems to ensure that the reactor core is protected after any of the following accidents:

- a) Loss of coolant accident (LOCA);
- b) Rod ejection accident (REA);
- c) Steam generator tube rupture (SGTR); and
- d) Main steam line break (MSLB).

TS 3.5.2 is applicable in MODES 1, 2, and 3.

TS 3.5.3, ECCS - Shutdown, identifies that one ECCS low pressure injection (LPI) subsystem shall remain operable. One of the two ECCS LPI subsystems is required to ensure sufficient ECCS flow is available to the core following a DBA. This subsystem includes an LPI pump, a decay heat cooler, and supporting piping, valves, instrumentation, and controls. TS 3.5.3 is applicable in MODE 4.

TS 3.5.4, Borated Water Storage Tank (BWST), identifies that the BWST shall remain operable. The BWST supplies borated water for ECCS and containment spray pump operation. It also supplies borated water to the refuel canal for refueling operations. TS 3.5.4 is applicable in MODES 1, 2, 3, and 4.

All TS in Section 3.5 do not apply with the reactor defueled. Therefore, these emergency core cooling system requirements do not apply and are being proposed for deletion.

Summary:

The above TS are related to assuring the appropriate functional capability of ECCS required for mitigation of design basis accidents only when the reactor is in Modes 1 through 4. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which address the ECCS, will no longer be applicable. Based on the above, the proposed deletion of all TS in Section 3.5 is acceptable, and the deletion of

these TS will have no impact on continued safe maintenance of the facility. The corresponding TS bases will also be deleted.

| <b>TS Section 3.6 Containment Systems</b>   |                    |
|---|--------------------|
| Current DBNPS TS  | Proposed DBNPS TS  |
| TS 3.6.1 – Containment  | TS 3.6.1 – Deleted |
| TS 3.6.2 – Containment Air Locks  | TS 3.6.2 – Deleted |
| TS 3.6.3 – Containment Isolation Valves   | TS 3.6.3 – Deleted |
| TS 3.6.4 – Containment Pressure   | TS 3.6.4 – Deleted |
| TS 3.6.5 – Containment Air Temperature  | TS 3.6.5 – Deleted |
| TS 3.6.6 – Containment Spray and Air Cooling Systems  | TS 3.6.6 – Deleted |
| TS 3.6.7 – Trisodium Phosphate Dodecahydrate (TSP) Storage  | TS 3.6.7 – Deleted |
| <b>Basis</b>  |                    |
| <p>TS Section 3.6, Containment Systems, contains LCOs that provide for appropriate functional capability of containment systems required for safe operation of the facility.</p> <p>The TS listed below do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS bases are also proposed for deletion to reflect this change.</p> <p>TS 3.6.1, Containment, identifies that containment shall be operable. The containment vessel, including all penetrations, is designed to withstand a loss-of-coolant accident and confine a postulated release of radioactive material. It is a cylindrical steel vessel, completely enclosed by a reinforced concrete shield building. TS 3.6.1 is applicable in MODES 1, 2, 3, and 4.</p> <p>TS 3.6.2, Containment Air Locks, identifies that two containment air locks shall be operable. The containment air locks form part of the containment pressure boundary and provide a means for personnel access. They are designed and tested to ensure they can withstand a pressure in excess of the maximum expected pressure following a DBA. TS 3.6.2 is applicable in MODES 1, 2, 3, and 4.</p> |                    |

TS 3.6.3, Containment Isolation Valves, identifies that each containment isolation valve shall be operable. The containment isolation valves are part of the containment pressure boundary and make up the containment isolation system. These consist of two barriers in series so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds the limits. TS 3.6.3 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.4, Containment Pressure, identifies that containment pressure shall be greater than or equal to -14 inches water gauge and less than or equal to +25 inches water gauge. Maintaining containment pressure within its limits ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. TS 3.6.4 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.5, Containment Air Temperature, identifies that containment average temperature shall be less than or equal to 120°F. Maintaining containment temperature within its limit ensures that, in the event of a DBA, the temperature profile is bounded and required safety related equipment will continue to perform its function. TS 3.6.5 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.6, Containment Spray and Air Cooling Systems, identifies that two containment spray trains and two containment air cooling trains shall be operable. Containment spray and containment air cooling (CAC) systems provide containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. The containment spray system consists of two separate, independent trains of equal capacity, each capable of meeting the design basis. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. The containment air cooling system consists of three containment cooling trains that draw air from the containment atmosphere and discharge into a common supply plenum. TS 3.6.6 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.7, Trisodium Phosphate Dodecahydrate (TSP) Storage identifies that the TSP storage baskets shall contain greater than or equal to 290 ft<sup>3</sup> of TSP. The TSP storage baskets are a subsystem of the containment spray system that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a DBA. During an accident such as a LOCA, the containment emergency sump will flood above the TSP storage baskets, dissolving the TSP and mixing it with the emergency sump water. This promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. TS 3.6.7 is applicable in MODES 1, 2, 3, and 4.

All TS in Section 3.6 do not apply with the reactor defueled. Therefore, these containment system requirements do not apply and are being proposed for deletion.

Summary:

All TS in Section 3.6 are related to assuring the appropriate functional capability of plant equipment associated with containment systems required for safe operation of the facility and accident mitigation only when the reactor is in MODES 1 through 4. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which only address containment systems, are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.6 is acceptable, and the deletion of these TS will have no impact on continued safe maintenance of the facility. The corresponding TS bases will also be deleted.

| <b>TS Section 3.7 Plant Systems</b>   |                          |
|---|--------------------------|
| <b>Current DBNPS TS</b>   | <b>Proposed DBNPS TS</b> |
| TS 3.7.1 – Main Steam Safety Valves (MSSVs)   | TS 3.7.1 – Deleted       |
| TS 3.7.2 – Main Steam Isolation Valves (MSIVs)  | TS 3.7.2 – Deleted       |
| TS 3.7.3 – Main Feedwater Stop Valves (MFSVs), Main Feedwater Control Valves (MFCVs), and associated Startup Feedwater Control Valves (SFCVs) | TS 3.7.3 – Deleted       |
| TS 3.7.4 – Turbine Stop Valves (TSVs)   | TS 3.7.4 – Deleted       |
| TS 3.7.5 – Emergency Feedwater (EFW)  | TS 3.7.5 – Deleted       |
| TS 3.7.6 – Condensate Storage Tanks (CSTs)  | TS 3.7.6 – Deleted       |
| TS 3.7.7 – Component Cooling Water (CCW) System   | TS 3.7.7 – Deleted       |
| TS 3.7.8 – Service Water System (SWS)   | TS 3.7.8 – Deleted       |
| TS 3.7.9 – Ultimate Heat Sink (UHS)   | TS 3.7.9 – Deleted       |
| TS 3.7.10 – Control Room Emergency Ventilation System (CREVS)   | TS 3.7.10 – Deleted      |
| TS 3.7.11 – Control Room Emergency Air Temperature Control System (CREATCS)   | TS 3.7.11 – Deleted      |

|   |                     |
|---|---------------------|
| TS 3.7.12 – Station Emergency Ventilation System (EVS)  | TS 3.7.12 – Deleted |
| TS 3.7.13 – Spent Fuel Pool Area Emergency Ventilation System (EVS)   | TS 3.7.13 – Deleted |
| TS 3.7.14 – Spent Fuel Pool Water Level   | TS 3.7.14 – Revised |
| TS 3.7.15 – Spent Fuel Pool Boron Concentration   | TS 3.7.15 – Revised |
| TS 3.7.16 – Spent Fuel Pool Storage   | TS 3.7.16 – Revised |
| TS 3.7.17 – Secondary Specific Activity   | TS 3.7.17 – Deleted |
| TS 3.7.18 – Steam Generator Level   | TS 3.7.18 – Deleted |
| <b>Basis</b>  |                     |
| <p>TS Section 3.7, Plant Systems, contains LCOs that provide for appropriate functional capability of plant equipment required for safe operation of the facility, including the plant being in a defueled condition.</p> <p>The following TS in Section 3.7 are being proposed for deletion: TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.8, TS 3.7.9, TS 3.7.10, TS 3.7.11, TS 3.7.12, TS 3.7.13, TS 3.7.17, and TS 3.7.18. The corresponding TS bases sections are also being proposed for deletion to reflect this change.</p> <p>TS being retained and revised are TS 3.7.14, TS 3.7.15, and TS 3.7.16 as further described below. The corresponding TS bases sections are also being proposed for revision to reflect this change.</p> <p>TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.8, TS 3.7.9, TS 3.7.11, TS 3.7.12, TS 3.7.17, and TS 3.7.18 do not apply with the reactor defueled.</p> <p>TS 3.7.1, Main Steam Safety Valves (MSSVs), identifies that the MSIVs shall be OPERABLE in accordance with Table 3.7.1-1. The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against over-pressurizing the RCPB by providing a heat sink for removal of energy from the RCS if the preferred heat sink, provided by the condenser and circulating water system, is not available. TS 3.7.1 is applicable in MODES 1, 2, and 3.</p> <p>TS 3.7.2, Main Steam Isolation Valves (MSIVs), identifies that two MSIVs shall be operable. MSIVs isolate steam flow from the secondary side of the steam generators following a main steam or feedwater line break. MSIV closure terminates flow from the unaffected (intact) steam generator. One MSIV is located in each main steam line</p> |                     |



outside of, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater pump turbine's steam supply to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, turbine bypass system, and other auxiliary steam supplies from the steam generators. TS 3.7.2 is applicable in MODE 1, and in MODES 2 and 3 except when all MSIVs are closed.

TS 3.7.3, Main Feedwater Stop Valves (MFSVs), Main Feedwater Control Valves (MFCVs), and associated Startup Feedwater Control Valves (SFCVs), identifies that two MFSVs and associated SFCVs shall be operable. The main feedwater isolation valves (MFIVs) for each steam generator consist of the MFSVs, MFCVs, and the SFCVs. The MFIVs isolate main feedwater flow to the secondary side of the steam generators following a high energy line break (HELB). Closure of the MFIVs terminates flow to both steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the feedwater lines downstream of the MFIVs will be mitigated by their closure. Closing the MFIVs and associated bypass valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for MSLBs or FWLBs inside containment and reducing the cooldown effects for MSLBs. TS 3.7.3 is applicable in MODES 1, 2, and 3 except when all MFSVs, MFCVs, and associated SFCVs are closed or isolated by a closed manual valve.

TS 3.7.4, Turbine Stop Valves (TSVs), identifies that four TSVs shall be operable. The TSVs are designed to quickly shut off steam flow to the turbine and prevent turbine overspeed under emergency conditions. TSV closure also terminates flow from the unaffected (intact) steam generator following an MSLB. TS 3.7.4 is applicable in MODE 1, and in MODES 2 and 3 except when all TSVs are closed.

TS 3.7.5, Emergency Feedwater (EFW), identifies that three EFW trains (consisting of two auxiliary feedwater (AFW) trains and the motor driven feedwater pump train) shall be operable. The EFW system provides a safety related source of feedwater to the secondary side of the steam generators in the event of a loss of normal feedwater flow to remove reactor decay heat. TS 3.7.5 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.6, Condensate Storage Tanks (CSTs), identifies that the CSTs shall be operable. The two CSTs provide the primary source of water to the steam generators for removing decay and sensible heat from the RCS. The CSTs provide a passive flow of water, by gravity, to the AFW System and the motor driven feedwater pump when aligned to the AFW mode (LCO 3.7.5, "Emergency Feedwater (EFW)"). TS 3.7.6 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.7, Component Cooling Water (CCW) System, identifies that two CCW loops shall be operable. The CCW system provides a heat sink for the removal of process and



operating heat from safety related components during a DBA or transient. During normal operation, the CCW system also provides this function for various nonessential components, as well as the spent fuel pool. The CCW system serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the service water system, and thus to the environment. TS 3.7.7 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.8, Service Water System (SWS), identifies that two SWS loops shall be operable. The SWS provides a heat sink for the removal of process and operating heat from safety related components during a DBA or transient. During normal operation and normal shutdown, the SWS also provides this function for various safety related and non-safety related components. The safety related portion is covered by this LCO. TS 3.7.8 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.9, Ultimate Heat Sink (UHS), identifies that the UHS shall be operable. The UHS (Lake Erie) provides a heat sink for process and operating heat from safety related components during a transient or accident as well as during normal operation. This is done utilizing the SWS. TS 3.7.9 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.11, Control Room Emergency Air Temperature Control System (CREATCS), identifies that two CREATCS shall be operable. The CREATCS provides temperature control for the control room following isolation of the control room. The CREATCS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, valves or dampers, and instrumentation also form part of the system. Two redundant air cooled condensing units are provided as a backup to the water cooled condensing unit. Both the water cooled and air cooled condensing units must be OPERABLE for the CREATCS to be OPERABLE. During emergency operation, the CREATCS maintains the temperature less than or equal to 110°F in the control room. The CREATCS is a subsystem providing air temperature control for the control room. TS 3.7.11 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.12, Station Emergency Ventilation System (EVS), identifies that two station EVS trains shall be operable. The function of the EVS is to collect and process potential leakage from the containment vessel to minimize environmental activity levels resulting from all sources of containment leakage following a LOCA. TS 3.7.12 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.17, Secondary Specific Activity, identifies a limit on secondary coolant specific activity during power operation. A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents. TS 3.7.17 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.18, Steam Generator Level, identifies the water level requirements for each steam generator. Steam generator water inventory is maintained large enough to provide adequate primary to secondary heat transfer. Mass inventory and indicated water level in the steam generator increases with load as the length of the four heat transfer regions within the steam generator vary. Inventory is controlled indirectly as a function of power and maintenance of a constant average primary system temperature by the feedwater controls in the integrated control system. TS 3.7.18 is applicable in MODES 1, 2, and 3.

The TS listed above do not apply with the reactor defueled; therefore, these plant systems requirements do not apply and are being proposed for deletion. The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 4. Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the TS listed in the previous paragraphs, which address plant systems, are no longer applicable. Based on the above, the proposed deletion of the above described TS in Section 3.7 is acceptable. The corresponding TS bases will also be deleted.

The following TS in Section 3.7, (TS 3.7.10 and TS 3.7.13) are also being proposed for deletion:

TS 3.7.10, Control Room Emergency Ventilation System (CREVS), identifies two CREVS trains shall be operable. The CREVS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The CREVS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREVS train consists of a roughing filter, a HEPA filter, a charcoal adsorber for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system. The CREVS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a design basis accident (DBA), without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA. During movement of irradiated fuel assemblies, the CRE boundary must be OPERABLE to cope with a release due to a fuel handling accident.

The design basis accidents and transients analyzed in UFSAR Chapter 15 are no longer applicable in the permanently defueled condition, with the exception of the FHA outside containment, the WGDTR, and the external causes. A description of the FHA, the WGDTR, and the external causes analyses for the permanently defueled condition was previously provided in this amendment request.

The FHA analysis shows that the dose consequences are acceptable without relying on any active components to remain functional (including the CREVS) during and following the event. The FHA analysis results determined that at least 95 days of irradiated fuel decay time after reactor shutdown and compliance with the spent fuel pool water level requirements of TS 3.7.14 are required for the control room, EAB, and LPZ doses to remain below the acceptance criteria. As such, the CREVS is not required to ensure that the accident dose at the site boundary will remain well below the 10 CFR 100 limits and the control room dose will be within the 10 CFR 50, GDC 19 limits. Consequently, the CREVS is not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA after the required 95 day after shutdown decay period has elapsed.

The requirement for the CREVS was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii), which states that TS limiting conditions for operation must be established for structures, systems, and components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the CREVS is no longer relied upon for accident mitigation, it is also not required during movement of irradiated fuel assemblies for mitigation of a potential FHA after 95 days of decay time following the permanent shutdown and compliance with the spent fuel pool water level requirements of TS 3.7.14. 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 the design basis accident that is credible with the unit permanently defueled. As such, the requirement for the CREVS is being deleted because there are no design basis events that rely on the CREVS for mitigation, and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) no longer apply.

Based on the above, the proposed deletion of TS 3.7.10 is acceptable. The corresponding TS bases will also be deleted.

TS 3.7.13, Spent Fuel Pool Area Emergency Ventilation System (EVS), identifies that two spent fuel pool area EVS trains shall be operable. The spent fuel pool area EVS provides negative pressure in the spent fuel pool area, and filters airborne radioactive particulates and iodines from the area of the spent fuel pool following a fuel handling accident. With the containment equipment hatch open, the spent fuel pool area negative pressure boundary extends to include the inside of the containment pressure vessel.

The spent fuel pool area EVS consists of portions of the normal fuel handling area ventilation system (FHAVS), the station EVS, ductwork bypasses, and dampers. The portion of the normal FHAVS used by the spent fuel pool area EVS consists of ducting between the spent fuel pool and the normal FHAVS exhaust fans or dampers, and redundant radiation detectors installed close to the suction end of the FHAVS exhaust fan ducting. The portion of the station EVS used by the spent fuel pool area EVS consists of two independent, redundant trains. Each train consists of a prefilter, HEPA filter, activated charcoal adsorber section for removal of gaseous activity (principally iodines), and fan. Ductwork, valves or dampers, and instrumentation also form part of the system. Two dampers are installed in series in the ductwork between the FHAVS and the station EVS to provide isolation of the station EVS from the FHAVS on a safety features actuation signal. These dampers are normally open. The station EVS is the subject of LCO 3.7.12, "Station Emergency Ventilation System (EVS)." A ductwork bypass with redundant dampers connects the FHAVS to the station EVS.

During normal operation, the exhaust from the fuel handling area is passed through the FHAVS exhaust filter and is discharged through the station vent stack. In the event of a fuel handling accident, the radiation detectors (one per train), located at the suction of the FHAVS exhaust fan ducting, send signals to isolate the FHAVS supply and exhaust fans and ductwork, open the redundant dampers in the bypass ductwork, and start the station EVS fans. The station EVS fans pull the air from the fuel handling area, creating a negative pressure, and discharge the filtered air to the station vent.

TS 3.7.13 is applicable during movement of irradiated fuel assemblies in the spent fuel pool building.

The design basis accidents and transients analyzed in UFSAR Chapter 15 are no longer applicable in the permanently defueled condition, with the exception of the FHA outside containment, the WGDTR, and the external causes. A description of the FHA, the WGDTR, and the external causes analyses for the permanently defueled condition was previously provided in this amendment request.

The FHA analysis shows that the dose consequences are acceptable without relying on any active components to remain functional (including the spent fuel pool area EVS system) during and following the event. The FHA analysis results determined that at least 95 days of irradiated fuel decay time after reactor shutdown and compliance with the spent fuel pool water level requirements of TS 3.7.14 are required for the control room, EAB, and LPZ doses to remain below the acceptance criteria. As such, the spent fuel pool EVS system is not required to ensure that the accident dose at the site boundary will remain well below the 10 CFR 100 limits and the control room dose will be within the 10 CFR 50, GDC 19 limits. Consequently, the spent fuel pool EVS system is not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA after the required 95 day after shutdown period has been met.

The requirement for the spent fuel pool EVS system was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii), which states that TS limiting conditions for operation must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the spent fuel pool EVS system is no longer relied upon for accident mitigation, it is also not required during movement of irradiated fuel assemblies for mitigation of a potential FHA. 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 the design basis accident that is credible with the unit permanently defueled. As such, the requirement for the spent fuel pool EVS system is being deleted because there are no design basis events that rely on the spent fuel pool EVS system for mitigation, and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) no longer apply.

Based on the above, the proposed deletion of TS 3.7.13 for the spent fuel pool EVS system is acceptable. The corresponding TS Bases will also be deleted.

The following TS in Section 3.7 (TS 3.7.14, TS 3.7.15, and TS 3.7.16) are proposed to be revised as follows:

TS 3.7.14, Spent Fuel Pool Water Level, identifies the spent fuel pool water level over the top of irradiated fuel assemblies seated in the storage racks. The minimum water level in the spent fuel pool meets the assumption 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.

TS 3.7.14 is applicable during movement of irradiated fuel assemblies in the spent fuel pool.

TS 3.7.14 is being retained in the permanently defueled TS with the following change. The Note in Required Action A.1 (LCO 3.0.3 is not applicable) is being deleted to conform to the deletion of TS LCO 3.0.3 described in the TS Section 3.0 above. With the deletion of LCO 3.0.3, this note is rendered moot.

Retaining TS 3.7.14, with the proposed change, continues to ensure appropriate requirements for spent fuel pool water level.

TS 3.7.15, Spent Fuel Pool Boron Concentration, identifies that the spent fuel pool boron concentration shall be greater than or equal to 630 ppm. Fuel assemblies are stored in the spent fuel pool racks in a mixed zone three region, checkerboard, or homogenous loading pattern in accordance with criteria based on initial enrichment and assembly burnup. The high density spent fuel pool storage racks in the SFP are designed to assure that the effective neutron multiplication factor,  $k_{eff}$ , is less than or equal to 0.95

with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water.

Three potential accident scenarios, misloaded fresh fuel assembly, mislocated fresh fuel assembly, and a dropped fuel assembly, were analyzed to determine the effect the accidents would have on the effective neutron multiplication factor,  $k_{eff}$ . The results of the analysis determined that a minimum boron concentration of 630 ppm in the SFP water is required to maintain  $k_{eff}$  less than or equal to 0.95 for the worst-case accident scenario (a 5.05 weight percent enriched fresh fuel assembly misloaded in a checkerboard pattern). The minimum boron concentration value of 630 ppm bounds all analyzed potential accident scenarios.

TS 3.7.15 applies when fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

TS 3.7.15 is being retained in the permanently defueled TS with the following change. The Note in Required Action A (LCO 3.0.3 is not applicable) is being deleted to conform to the deletion of TS LCO 3.0.3 described in the TS Section 3.0 above. With the deletion of LCO 3.0.3, this note is rendered moot.

Retaining TS 3.7.15, with the proposed change, continues to ensure appropriate requirements for spent fuel pool boron concentration.

TS 3.7.16, Spent Fuel Pool Storage, identifies the restrictions on the placement of fuel assemblies in accordance with the criteria shown in Figure 3.7.16-1.

The spent fuel storage facility is designed for noncriticality by use of adequate spacing. A neutron absorber is attached to all four sides of each cell. In addition, there is a gap between individual racks and between the peripheral racks and the pool walls. These gaps form flux traps that reduces neutron movement between fuel assemblies in adjacent racks. Loading patterns maintain  $k_{eff}$  less than 0.95 for fuel assemblies with initial nominal enrichments less than or equal to 5.05 weight percent uranium-235, assuming the spent fuel pool water is unborated.

TS 3.7.16 applies whenever any fuel assembly is stored in the spent fuel pool.

TS 3.7.16 is being retained in the permanently defueled TS with the following change. The Note in Required Action A.1 (LCO 3.0.3 is not applicable) is being deleted to conform to the deletion of TS LCO 3.0.3 described in the TS Section 3.0 above. With the deletion of LCO 3.0.3, this note is rendered moot.

Retaining TS 3.7.16, with the proposed change, continues to ensure appropriate requirements for storing fuel assemblies in the spent fuel pool storage.



**Summary:**

TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.8, TS 3.7.9, TS 3.7.11, TS 3.7.12, TS 3.7.17, and TS 3.7.18 will not apply once the reactor is permanently defueled and are not needed for a permanently shutdown and defueled condition. As such, they may be deleted with no impact on continued safe maintenance of the facility.

TS 3.7.10 and TS 3.7.13 are not needed for accident mitigation in the permanently defueled condition. As such, they may be deleted with no impact on continued safe maintenance of the facility.

TS 3.7.14, TS 3.7.15 and TS 3.7.16 will remain applicable with the reactor permanently defueled. As such, they are being retained and revised to reflect a permanently defueled condition.

The corresponding TS bases sections are also being deleted or revised to reflect this change, as appropriate.

| <b>TS Section 3.8 Electrical Power Systems</b>          |                          |
|---|--------------------------|
| <b>Current DBNPS TS</b>                                 | <b>Proposed DBNPS TS</b> |
| TS 3.8.1 – [Alternating current] AC Sources - Operating | TS 3.8.1 – Deleted       |
| TS 3.8.2 – AC Sources - Shutdown                        | TS 3.8.2 – Deleted       |
| TS 3.8.3 – Diesel Fuel Oil, Lube Oil, and Starting Air  | TS 3.8.3 – Deleted       |
| TS 3.8.4 – [Direct current] DC Sources - Operating      | TS 3.8.4 – Deleted       |
| TS 3.8.5 – DC Sources - Shutdown                        | TS 3.8.5 – Deleted       |
| TS 3.8.6 – Battery Parameters                           | TS 3.8.6 – Deleted       |
| TS 3.8.7 – Inverters - Operating                        | TS 3.8.7 – Deleted       |
| TS 3.8.8 – Inverters - Shutdown                         | TS 3.8.8 – Deleted       |
| TS 3.8.9 – Distribution Systems - Operating             | TS 3.8.9 – Deleted       |
| TS 3.8.10 – Distribution Systems - Shutdown             | TS 3.8.10 – Deleted      |



| Basis  |
|--|
| <p>TS Section 3.8, Electrical Power Systems, contains LCOs that provide for appropriate functional capability of plant electrical equipment required for safe operation of the facility.</p> <p>The TS listed below (TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9) do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS Bases are also proposed for deletion to reflect this change.</p> <p>TS 3.8.1, AC Sources – Operating, identifies the AC electrical power sources that shall be operable. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF Systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.1 is applicable in MODES 1, 2, 3, and 4.</p> <p>TS 3.8.4, DC Sources – Operating, identifies that two trains of DC electrical power shall be operable. The DC electrical power sources (Train 1 and Train 2), each train consisting of two batteries, one battery charger for each battery, the corresponding control equipment, and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. Loss of any train DC electrical power source does not prevent the minimum safety function from being performed. TS 3.8.4 is applicable in MODES 1, 2, 3, and 4.</p> <p>TS 3.8.7, Inverters – Operating, identifies that two trains of inverters shall be operable. The inverter provides an uninterruptible power source for the instrumentation and controls for the RPS and the SFAS. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and SFAS instrumentation and controls so that the fuel, reactor coolant system, and containment design limits are not exceeded. TS 3.8.7 is applicable in MODES 1, 2, 3, and 4.</p> <p>TS 3.8.9, Distribution Systems – Operating, identifies that two trains of AC, DC, and AC vital bus electrical power distribution subsystems shall be operable. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system, and containment design limits are not exceeded. TS 3.8.9 is applicable in MODES 1, 2, 3, and 4.</p> <p>TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 are related to assuring the appropriate functional capability of plant equipment required for safe operation of the facility only when the reactor is in MODES 1 through 4. Once both the certification of permanent</p> |

cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 are no longer applicable and are being proposed for deletion. Based on the above, the proposed deletion of TS related to these systems is acceptable.

TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10 are also being proposed for deletion.

TS 3.8.2, AC Sources – Shutdown, identifies the AC electrical power sources that shall be operable during MODES 5 and 6, and during movement of irradiated fuel.

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the technical specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst-case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems. TS 3.8.2 is applicable in MODES 5 and 6, and during movement of irradiated fuel assemblies.

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, TS 3.8.2 will no longer be needed for assuring the appropriate functional capability of the AC sources for safe operation of the facility when the reactor is in MODES 5 and 6. The only remaining TS 3.8.2 Applicability requirement for AC sources is during movement of irradiated fuel

assemblies. However, the updated applicable accident analyses show that the dose consequences are acceptable without relying on any active components to remain functional (including the AC sources) during and following the event. Therefore, TS 3.8.2 no longer applies and is being proposed for deletion.

The remaining applicable design basis accidents and transients analyzed in UFSAR Chapter 15 were discussed previously in this proposed amendment, including a description of each accident with the potential to result in a radiological release. The accident analyses show that the dose consequences are acceptable without relying on any active components to remain functional during and following the event.

The requirement for AC sources was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii), which states that TS limiting conditions for operation must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The FHA is the applicable design basis accident related to the TS requirement for functional capability of AC sources (offsite power and EDGs) during the TS specified condition of "During movement of irradiated fuel assemblies." Because the updated FHA analysis does not rely on normal or emergency power for accident mitigation after 95 days of decay time following the permanent shutdown of the reactor (including any need for providing airborne radiological protection), the AC sources are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. Therefore, during movement of fuel assemblies in the fuel storage pool, 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 the design basis accident that is credible with the unit permanently defueled. As such, the requirement for AC sources is being deleted because there are no design basis events that rely on AC sources for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) no longer apply.

With the reactor permanently defueled, irradiated fuel is stored either in the DFSF or in the SFP. The DFSF is a passive system that does not rely on electrical power for heat transfer. Since there is a large capacity for heat absorption in the SFP, active system components are not redundant. The DBNPS UFSAR Section 9.1.3.3.3 details the timing of a boiloff of the SFP following the unlikely failure of the SFP cooling system. The conclusion is that there is sufficient time for remedial actions to maintain or restore the SFP water level.

The existing requirement for a qualified offsite circuit is based on its need to be capable of maintaining rated frequency and voltage, accepting required loads during an accident, and capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, Distribution Systems - Shutdown. Because the requirement for ESF equipment no longer exists upon permanent defueling (as justified in the associated sections of this proposed amendment), there is no longer a need for essential buses. Since the AC

electrical power distribution subsystems are no longer required to power ESF equipment, TS 3.8.10 is being deleted, as described in the corresponding section below.

With no need for a qualified offsite circuit to be capable of supplying loads during an accident while connected to the essential buses, and no need for a DG, there is no longer a need for TS 3.8.2. Therefore, TS 3.8.2 is being deleted in its entirety.

TS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, identifies that stored diesel fuel oil, lube oil, and starting air subsystems must be within limits stated in the specification conditions. For proper operation of the EDGs, it is necessary to ensure sufficient quantity and proper quality of the fuel oil as well as sufficient quantity of lube oil. Stored diesel fuel oil is required to have sufficient supply for seven days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lube oil supply must be available to ensure the capability to operate at full load for seven days. This requirement, in conjunction with an ability to obtain replacement supplies within seven days, supports the availability of EDGs required to shut down the reactor and to maintain it in a safe condition for an AOO or a postulated DBA with loss of offsite power. Each EDG has an air start system with two air start receivers per subsystem, and each air start receiver has adequate capacity for five successive start attempts on the EDG without recharging the air start receiver. TS 3.8.3 is applicable when associated EDG is required to be OPERABLE.

The AC sources (TS 3.8.1 and TS 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and starting air support TS 3.8.1 and TS 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE. As such TS 3.8.3 is applicable when the associated DG is required to be OPERABLE.

TS 3.8.3 is required to support the DG requirements of TS 3.8.1 and TS 3.8.2. With the deletion of TS 3.8.1 and TS 3.8.2, the requirements of TS 3.8.3 are no longer applicable. Therefore, TS 3.8.3 is being proposed for deletion.

As discussed in the justification for deleting TS 3.8.2 above, the requirement for EDGs is being deleted from the TS because there are no design basis accidents or transients applicable to the facility in a permanently defueled condition that rely on the EDGs for mitigation. Since TS 3.8.3 exists solely to support the EDG requirements of TS 3.8.1 and TS 3.8.2, the elimination of the need for DGs also obviates the need for their support systems. As such, TS 3.8.3 may be deleted. Based on the above, the proposed deletion of TS 3.8.3 for fuel oil, lube oil, and starting air parameters is acceptable.

TS 3.8.5, DC Sources – Shutdown, identifies that DC electrical subsystems must be OPERABLE to support the DC electrical distribution subsystems required by LCO 3.8.10. The DC electrical power system provides normal and emergency DC electrical power for the EDGs, emergency auxiliaries, and control and switching during all MODES

of operation. One train of the DC electrical power sources consisting of two batteries, one battery charger per battery, the corresponding control equipment, and interconnecting cabling within the train is required to be OPERABLE to support one train of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (fuel handling accidents). TS 3.8.5 is applicable during MODES 5 and 6, and during movement of irradiated fuel assemblies.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, TS 3.8.5 will no longer be needed for assuring the appropriate functional capability of the DC sources for safe operation of the facility when the reactor is in MODES 5 or 6 or during the movement of irradiated fuel assemblies. Also, the remaining applicable accident analyses do not rely on DC sources for accident mitigation. Consequently, DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential accident. Therefore, TS 3.8.5 is being proposed for deletion.

The remaining applicable design basis accidents and transients analyzed in UFSAR Chapter 15 were discussed previously in this proposed amendment, including a description of each accident with the potential to result in a radiological release. Because the post defueling accident analyses do not rely on DC sources for accident mitigation (dose consequences after 95 days of decay time following permanent shutdown are acceptable without relying on any active components to remain functional during and following the event), DC sources are therefore not required for accident mitigation. Consequently, DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA after 95 days following the permanent shutdown. Thus, the requirement for DC sources is being deleted.

The requirement for DC sources was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii), which states that TS limiting conditions for operation 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. Because the FHA analysis does not rely on DC sources for accident mitigation, DC sources are therefore not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. 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 the DBA that is credible with the unit permanently defueled. As such, the requirement for DC sources is being deleted because there are no design basis accidents or transients analyzed in UFSAR Chapter 15 that are still applicable to a facility in a permanently defueled condition that rely on DC sources for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) no longer apply.

TS 3.8.6, Battery Parameters, identifies the Train 1 and Train 2 battery parameter limits. This LCO delineates the limits on battery float current as well as cell electrolyte temperature, level, and float voltage for the DC electrical power source batteries. In addition to the limitations of this specification, the Battery Monitoring and Maintenance Program also implements a program specified in Specification 5.5.16 for monitoring various battery parameters. Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. TS 3.8.6 is applicable when associated DC electrical power sources are required to be OPERABLE.

Battery parameters are required solely for the support of the associated DC electrical power subsystems (per TS 3.8.4 and TS 3.8.5). Therefore, battery parameter limits are only required (and TS 3.8.6 is only applicable) when the DC electrical power source is required to be OPERABLE. As TS 3.8.4 and TS 3.8.5 are being proposed for deletion, TS 3.8.6 is also being proposed for deletion.

As discussed in the justification for deleting TS 3.8.5 above, the requirement for DC sources is being deleted from the TS because there are no design basis accidents and transients analyzed in UFSAR Chapter 15 still applicable to a facility in a permanently defueled condition that rely on the DC sources for mitigation. Since TS 3.8.6 exists solely to support the DC source requirements of TS 3.8.4 and TS 3.8.5, the elimination of the need for DC sources also obviates the need for their support systems. As such, TS 3.8.6 may be deleted. Based on the above, the proposed deletion of TS 3.8.6 for battery parameters is acceptable.

TS 3.8.8, Inverters – Shutdown, identifies that one inverter shall be OPERABLE to support the 120 volts alternating current (VAC) vital electrical distribution subsystem required by LCO 3.8.10, "Distribution Systems – Shutdown." The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and SFAS instrumentation and controls so that the fuel, RCS, and containment

design limits are not exceeded. TS 3.8.8 is applicable during MODES 5 and 6, and during movement of irradiated fuel assemblies.

The OPERABILITY of one inverter to a required 120 VAC vital bus during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, TS 3.8.8 will no longer be needed for assuring the appropriate functional capability of the inverters for safe operation of the facility when the reactor is in MODES 5 or 6 or during the movement of irradiated fuel assemblies. Also, the updated applicable accident analyses do not rely on the inverters for accident mitigation. Consequently, the inverters are not needed during movement of irradiated fuel assemblies for mitigation of a potential accident. Therefore, TS 3.8.8 is being proposed for deletion.

The requirement for inverters was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii), which states that TS limiting conditions for operation must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis, after 95 days of decay time following reactor shutdown, does not rely on inverters for accident mitigation, the inverters are therefore not required during movement of irradiated fuel assemblies for mitigation of a potential FHA. 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 the design basis accident that is credible with the unit permanently defueled. As such, the requirement for inverters is being deleted because there are no design basis accidents or transients analyzed in UFSAR Chapter 15 still applicable to a facility in a permanently defueled condition that rely on the inverters for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) no longer apply. Based on the above, the proposed deletion of TS 3.8.8 for inverters is acceptable.

TS 3.8.10, Distribution Systems – Shutdown, requires that the necessary portion of AC, DC, and AC Vital bus electrical power distribution subsystems be OPERABLE to support equipment required to be OPERABLE. The AC, DC, and AC vital bus electrical power



distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.10 is applicable in MODES 5 and 6 and during movement of irradiated fuel assemblies.

The OPERABILITY of the minimum AC, DC, and AC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

TS 3.8.10 explicitly requires energization of the portions of the electrical power distribution system necessary to support OPERABILITY of required systems, equipment, and components. Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (for example, fuel handling accidents).

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.10 is no longer needed for assuring the appropriate functional capability of the electrical distribution systems for safe operation of the facility when the reactor is in MODES 5 or 6 or during movement of irradiated fuel assemblies. Also, the updated applicable accident analyses do not rely on the electrical distribution systems for accident mitigation after 95 days of decay time following the permanent shutdown of the reactor. Consequently, the electrical distribution systems are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA after permanent shutdown. Therefore, TS 3.8.10 is being proposed for deletion.

The remaining applicable design basis accidents and transients analyzed in UFSAR Chapter 15 were discussed previously in this proposed amendment. A description of each accident with the potential to result in a radiological release was provided. The accident analysis shows that the dose consequences are acceptable after 95 days of decay time without relying on any active components to remain functional during and following the event.

The requirement for electrical distribution systems was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii), which states that TS limiting conditions for operation 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. Because the FHA analysis does not rely on electrical distribution systems for accident mitigation after 95 days of decay time, electrical distribution systems are therefore not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. 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 the design basis accident that is credible with the unit permanently defueled. As such, the requirement for electrical distribution systems is being deleted because there are no design basis events that rely on electrical distribution systems for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) no longer apply.

The existing requirement for the AC, DC, and AC vital bus electrical power distribution systems of TS 3.8.10 is to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. Because there is no longer a need for any ESF systems for accident mitigation, the requirements of TS 3.8.10 are no longer needed. Therefore, TS 3.8.10 is being deleted in its entirety.

Summary:

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 do not apply with the reactor defueled and are not needed for a permanently shutdown and defueled condition. As such, they may be deleted with no impact on continued safe maintenance of the facility.

TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10 are not needed for accident mitigation in the permanently defueled condition. As such, these specifications may be deleted with no impact on continued safe maintenance of the facility.

Based on the above, the proposed deletion of all TS in Section 3.8 is acceptable, and the deletion of these TS will have no impact on continued safe maintenance of the facility. The corresponding TS Bases will also be deleted.

| TS Section 3.9 Refueling Operations |                    |
|-------------------------------------|--------------------|
| Current DBNPS TS                    | Proposed DBNPS TS  |
| TS 3.9.1 – Boron Concentration      | TS 3.9.1 – Deleted |
| TS 3.9.2 – Nuclear Instrumentation  | TS 3.9.2 – Deleted |

|   |                    |
|---|--------------------|
| TS 3.9.3 – Decay Time   | TS 3.9.3 – Deleted |
| TS 3.9.4 – Decay Heat Removal (DHR) and Coolant Circulation – High Water Level  | TS 3.9.4 – Deleted |
| TS 3.9.5 – Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level   | TS 3.9.5 – Deleted |
| TS 3.9.6 – Refueling Canal Water Level  | TS 3.9.6 – Deleted |
| <b>Basis</b>  |                    |
| <p>TS Section 3.9, Refueling Operations, contains LCOs that provide for appropriate functional capability of parameters and equipment within containment that are required for mitigation of design basis accidents during refueling operations (moving fuel to or from the reactor core).</p> <p>The TS listed below do not apply once the reactor is permanently defueled; therefore, their corresponding LCOs (and associated SRs) are proposed to be deleted. The corresponding TS bases are also proposed for deletion to reflect this change.</p> <p>TS 3.9.1, Boron Concentration, identifies that boron concentrations in the RCS and refueling canal be maintained within the limit specified in the COLR. The limit on the boron concentrations of the RCS and the refueling canal during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of the volumes having direct access to the reactor core during refueling. The boron concentration limit specified in the COLR ensures a core <math>k_{eff}</math> of less than or equal to 0.95 is maintained during fuel handling operations with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. TS 3.9.1 is applicable in MODE 6.</p> <p>TS 3.9.2, Nuclear Instrumentation, identifies that two source range neutron flux monitors shall be OPERABLE. Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity, such as by a boron dilution accident. TS 3.9.2 is applicable in MODE 6.</p> <p>TS 3.9.3, Decay Time, identifies that the reactor should be subcritical for greater than or equal to 72 hours. This ensures sufficient time has elapsed to allow the radioactive decay of the short lived fission products. Prior to movement of irradiated fuel assemblies within the reactor vessel, the reactor must be subcritical for greater than or equal to 72 hours. This time period is an initial assumption of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Reference 100). The minimum time period of 72 hours ensures sufficient time has elapsed to allow the radioactive decay of the short-lived fission products, which helps ensure that the offsite doses during a fuel handling accident will be within the 10 CFR 100 limits. TS 3.9.3 is</p> |                    |

applicable during movement of irradiated fuel assemblies within the reactor pressure vessel.

TS 3.9.4, Decay Heat Removal (DHR) and Coolant Circulation – High Water Level, identifies that one DHR loop shall be OPERABLE and in operation. The DHR system in MODE 6 removes decay heat and sensible heat from the RCS, provides mixing of borated coolant, provides sufficient coolant circulation to minimize the effects of a boron dilution accident, and prevents boron stratification. One train of the DHR System is required to be operational in MODE 6, with the water level greater than or equal to 23 feet (ft) above the top of the reactor vessel flange. Only one DHR loop is required for decay heat removal in MODE 6, with a water level greater than or equal to 23 ft above the top of the reactor vessel flange. Only one DHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay heat removal capability. TS 3.9.4 is applicable in MODE 6 with the water level greater than or equal to 23 ft above the top of the reactor vessel flange.

TS 3.9.5, Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level, identifies that two DHR loops shall be OPERABLE and one DHR loop shall be in operation. The DHR system in MODE 6 removes decay heat and sensible heat from the RCS, provides mixing of borated coolant, provides sufficient coolant circulation to minimize the effects of a boron dilution accident, and prevents boron stratification. In MODE 6, with the water level less than 23 ft above the top of the reactor vessel flange, two independent DHR loops must be OPERABLE. Additionally, one DHR loop must be in operation to provide removal of decay heat and mixing of borated coolant to minimize the possibility of criticality. TS 3.9.5 is applicable in MODE 6 with the water level less than 23 ft above the top of reactor vessel flange.

TS 3.9.6, Refueling Canal Water Level, identifies that the refueling canal water level shall be maintained greater than or equal to 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the reactor vessel, the refueling canal, the fuel transfer canal, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident inside containment. With a minimum water level of 23 ft, and a minimum decay time of 72 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident inside containment is adequately captured by the water, and offsite doses are maintained within acceptable limits as provided by 10 CFR 100. TS 3.9.6 is applicable during movement of irradiated fuel assemblies within containment.

All TS in Section 3.9 do not apply with the reactor defueled. Therefore, these refueling operations requirements do not apply and are being proposed for deletion.

Summary:

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions

required for safe refueling operation of the facility only when the reactor is in MODE 6 or during movement of fuel assemblies within containment. However, once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2), which thereby precludes entry into MODE 6. The prohibition on placing fuel in the reactor vessel also precludes movement of fuel within containment. Therefore, the TS listed in the previous paragraphs, which only address these specific plant systems, control of process variables, design features, or operating restrictions are no longer applicable. Therefore, they may be deleted with no impact on continued safe maintenance of the facility. The corresponding TS bases will also be deleted.

| <b>TS Section 4.0 Design Features</b>  |                   |
|--|-------------------|
| Current DBNPS TS   | Proposed DBNPS TS |
| TS 4.2 – Reactor Core  | TS 4.2 – Deleted  |
| TS 4.3 – Fuel Storage  | TS 4.3 – Revised  |
| <b>Basis</b>   |                   |
| <p>TS Section 4.0, Design Features, contains descriptions and requirements for those features of the facility such as materials of construction and geometric arrangements which, if altered or modified, would have a significant effect on safety and are not covered in the previous sections of the TS. This section does not contain applicability requirements. As such, all parts of this section can be conservatively defined as being applicable at all times. There are no corresponding TS bases sections associated with this TS section.</p> <p>TS 4.2 does not apply once the reactor is permanently defueled; therefore, it is proposed to be deleted.</p> <p>TS 4.2, Reactor Core, provides a description and requirements regarding the reactor core fuel assemblies and control rod assemblies. TS 4.2 contains requirements only associated with the reactor core, which can no longer be used following submittal of the certifications required by 10 CFR 50.82(a). Therefore, TS 4.2 is not needed for a permanently defueled condition. 10 CFR 50.82(a)(2) prohibits FENOC from operating the plant or placing fuel in the reactor vessel. Therefore, TS 4.2 is no longer applicable. As such, TS 4.2 may be deleted with no impact on continued safe maintenance of the facility.</p> <p>TS 4.3 will be revised as follows:</p> |                   |

TS 4.3, Fuel Storage, provides a description and requirements regarding prevention of criticality of spent fuel, prevention of fuel storage pool drainage, and spent fuel capacity limitations. This TS section is being retained as-is in the permanently defueled TS, with the exception of TS 4.3.1.2. TS 4.3.1.2 provides a description and requirements regarding the design of the new fuel storage racks. This description is being proposed for deletion since new fuel is no longer stored onsite and License Condition 2.B.(3) is being revised to remove the allowance to receive new fuel. The design feature associated with the new fuel storage racks is no longer applicable and may be deleted.

Summary:

Once both the certification of permanent cessation of power operations and of permanent removal of fuel from the reactor vessel for DBNPS have been submitted in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the DBNPS 10 CFR 50 license will no longer authorize reactor operations or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Therefore, the proposed deletion of TS 4.2 and the proposed revision of TS 4.3 is acceptable and will have no impact on continued safe maintenance of the facility.

### **TS Section 5.5 Programs and Manuals**

This section provides a description and requirements regarding programs and manuals that are to be established, implemented, and maintained. TS 5.5 will remain applicable once the reactor is permanently defueled. As such, it is proposed to be retained and revised to reflect a permanently defueled condition as described below.

| Current DBNPS TS  | Basis for Change/Deletion  |
|---|--|
| TS 5.5.2 – Primary Coolant Sources Outside Containment  | <p>This program was established to provide controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable.</p> <p>This program is proposed for deletion. Once the plant is permanently shut down and defueled, there will no longer be any transient or accident conditions associated with primary coolant sources. Thus, TS 5.5.2 will not be retained.</p> |
| TS 5.5.4 – Reactor Vessel Internals Vent Valves Program | <p>This program was established to implement the testing of the reactor vessel internals vent valves every 24 months.</p> <p>The program is proposed for deletion because the DBNPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by</p>  |



|   |   |
|---|---|
|   | <p>10 CFR 50.82(a)(1) have been submitted. The reactor vessel internals vent valves will no longer be applicable in a permanently defueled condition. Thus, TS 5.5.4 will not be retained.</p>  |
| TS 5.5.5 – Allowable Operating Transient Cycles Program     | <p>This program was established to provide controls to track the cyclic and transient occurrences to ensure that the reactor vessel is maintained within the design limits.</p> <p>The program is proposed for deletion because the DBNPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. The reactor vessel will no longer be subjected to cycles or transients after permanent shutdown. Thus, TS 5.5.5 will not be retained.</p> |
| TS 5.5.6 – Reactor Coolant Pump Flywheel Inspection Program | <p>This program was established for the inspections of each reactor coolant pump flywheel.</p> <p>The program is proposed for deletion because the DBNPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. The reactor coolant pumps pertain only to reactor support systems that do not apply in a permanently defueled condition. Thus, TS 5.5.6 will not be retained.</p>  |
| TS 5.5.8 – Steam Generator (SG) Program                     | <p>This program was established to ensure that SG tube integrity is maintained.</p> <p>The program is proposed for deletion because the DBNPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. The steam generators pertain only to reactor support systems that do not apply in a permanently defueled condition. Thus, TS 5.5.8 will not be retained.</p>  |
| TS 5.5.9 – Secondary Water Chemistry Program                | <p>This program was established to provide controls for monitoring secondary water chemistry to inhibit SG tube degradation.</p> <p>The program is proposed for deletion because the DBNPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. The components that the program is established to protect are associated with reactor operation. Thus, TS 5.5.9 will not be retained.</p>  |



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|---|--|
| <p>TS 5.5.10 – Ventilation Filter Testing Program (VFTP)</p>    | <p>This program was established to implement the required testing of the filter ventilation systems for the EVS and control room ventilation systems.</p> <p>This program is proposed for deletion because the VFTP is no longer required in a permanently shut down and defueled condition. The accident analysis applicable to the permanently defueled condition does not rely on ventilation filters for accident mitigation. In addition, as previously discussed, TS 3.7.10 and TS 3.7.13 that provided the operability requirements for the CREVS and EVS are proposed to be eliminated. Thus, TS 5.5.10 will not be retained.</p>  |
| <p>TS 5.5.12 – Diesel Fuel Oil Testing Program</p>              | <p>This program was established to implement required testing of both new fuel oil and stored fuel oil used to supply the EDGs.</p> <p>This program is proposed for deletion because the technical specifications that provided operability requirements for the EDGs are proposed for elimination. Thus, TS 5.5.12 will not be retained.</p>  |
| <p>TS 5.5.14 – Safety Function Determination Program (SFDP)</p> | <p>This program was established to ensure loss of safety function is detected and appropriate actions taken.</p> <p>This program is proposed for deletion because the DBNPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. There will be no active SSCs required for accident mitigation with the permanent cessation of reactor operations and the permanent removal of the fuel from the reactor vessel. Therefore, the requirements of the SFDP, which directs cross-checks of multiple and redundant systems, no longer apply.</p>                        |
| <p>TS 5.5.15 – Containment Leakage Rate Testing Program</p>     | <p>This program was established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions.</p> <p>The program is proposed for deletion because the DBNPS Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. This program pertains only to verifying the operability of the containment systems. The requirements for containment systems (TS Section 3.6) are being deleted as described in this document. Thus, TS 5.5.15 will not be retained.</p> |

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| TS 5.5.16 – Battery Monitoring and Maintenance Program | <p>This program was established to provide safety-related battery restoration and maintenance.</p> <p>This program is proposed for deletion consistent with the deletion of the corresponding TS for DC electrical systems and associated batteries. The DBNPS accident analysis applicable to the permanently defueled condition does not rely on batteries for accident mitigation. Thus, TS 5.5.16 will not be retained.</p>   |
| TS 5.5.17 – Control Room Envelope Habitability Program | <p>This program was established and implemented to ensure that the CRE habitability was maintained such that, with an operable CREV system, the occupants of the CRE can control the reactor safely under normal and emergency conditions and maintain it in a safe condition following a radiological event, hazardous chemical release or a smoke challenge.</p> <p>This program is proposed for deletion. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted and after 95 days of decay following shut down, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. The design basis FHA was assessed for post-cessation of power operations in order to justify the elimination of TS requirements for the operability of control room ventilation systems. As previously discussed, TS 3.7.10 and TS 3.7.11 will not be retained in the PDTs. Thus, TS 5.5.17 will not be retained.</p> |

### TS Section 5.6 Reporting Requirements

This section provides a description and requirements regarding reports that are to be submitted in accordance with 10 CFR 50.4. TS 5.6 will remain applicable once the reactor is permanently defueled. As such, it is proposed to be retained and revised to reflect a permanently defueled condition as described below.

| Current DBNPS TS  | Basis for Change/Deletion  |
|---|--|
| TS 5.6.1 – Annual Radiological Environmental Operating Report | The term "unit" is typically associated with an operating reactor and is revised with the term "facility." This administrative change more appropriately represents the permanently shutdown and defueled condition. Proposed changes are shown in Attachment 1. |
| TS 5.6.2 – Radioactive Effluent Release Report                | The term "unit" is typically associated with an operating reactor and is revised with the term "facility." This administrative change more appropriately represents the permanently shutdown and defueled condition. Proposed changes are shown in Attachment 1. |

|   |  |
|---|--|
| TS 5.6.3 – CORE OPERATING LIMITS REPORT (COLR)  | According to TS 5.6.3, the COLR is established prior to each reload cycle or prior to any remaining portion of a reload cycle to document the specific limits associated with operating the reactor core and to ensure that the applicable limits of the safety analysis are met. This reporting requirement will not be retained in the PDTs because the Part 50 license will prohibit operation of the reactor or placement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Thus, the COLR does not apply in the permanently shut down and defueled condition.  |
| TS 5.6.4 – Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) | <p>RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates, shall be established and documented in the PTLR.</p> <p>This report is proposed for deletion. Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed, the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the RCS and the reactor support systems will no longer be used. The proposed deletion of this report is consistent with the proposed deletion of TS Section 3.4 described in this document. Thus, TS 5.6.4 will not be retained.</p> |
| TS 5.6.5 – Post-Accident Monitoring Report  | <p>This report was required by Condition B or F of LCO 3.3.17.</p> <p>This report is proposed for deletion. Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed, the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). This report pertains to an activity that does not apply in a permanently defueled condition. The proposed deletion of this report is consistent with the proposed deletion of TS 3.3.17 described in this document. Thus, TS 5.6.5 will not be retained.</p>   |
| TS 5.6.6 – Steam Generator Tube Inspection Report                                     | <p>This report was established in accordance with TS 5.5.8.</p> <p>This report is proposed for deletion. Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed, the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the SG</p>   |

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|--|--|
|  | tubes will not be subjected to the temperature and pressure effects that the SG Program and associated inspection report were put in place to monitor and assess. The proposed deletion of this report is consistent with the proposed deletion of TS 5.5.8 described in this document. Thus, TS 5.6.6 will not be retained.   |
| TS 5.6.7 – Remote Shutdown System Report | <p>This report was established in accordance with TS 3.3.18.</p> <p>This report is proposed for deletion. Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed, the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). This report pertains to an activity that does not apply in a permanently defueled condition. The proposed deletion of this report is consistent with the proposed deletion of TS 3.3.18 described in this document. Thus, TS 5.6.7 will not be retained.</p> |

### 3.0 REGULATORY EVALUATION

#### 3.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. FirstEnergy Nuclear Operating Company (FENOC) has determined that the proposed changes do not require any exemptions or relief from regulatory requirements.

##### 3.1.1 10 CFR 50.82, Termination of license

The portions of 10 CFR 50.82 providing the basis for this license amendment request (LAR) are:

- (a) For power reactor licensees—
  - (1) (i) 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 § 50.4(b)(8);
  - (ii) 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 § 50.4(b)(9) and;
  - (2) 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.

By letter dated April 25, 2018 (Reference 1), FENOC provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) of permanent cessation of power operations at DBNPS by May 31, 2020. After docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel. As a result, DBNPS will be authorized to only possess special nuclear material.

### **3.1.2 10 CFR 50.36 Technical Specifications**

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of the TS. Pursuant to 10 CFR 50.36, TS are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

10 CFR 50.36(c)(2)(ii)(A) through (D) provide criteria that require a licensee to include technical specifications for certain items. These criteria and their applicability to DBNPS in a permanently defueled condition are discussed below:

Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

When the reactor is permanently defueled, the reactor coolant system will no longer be in operation. Therefore, this criterion is no longer applicable.

Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The intent of this criterion is to capture process variables, design features, or operating conditions assumed in the safety analysis of an operating facility. Although the facility will no longer be in an operating mode, and the majority of the design basis events will no longer be applicable, there are still design basis events applicable to a plant authorized to handle, store, and possess nuclear fuel. The DBAs still applicable to DBNPS are discussed within this proposed amendment.

Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The intent of this criterion is to capture TS for those structures, systems, and components (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 design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that continue to apply to DBNPS, there are still design basis accidents that apply to a plant authorized only to handle, store, and possess nuclear fuel. The DBAs still applicable to DBNPS are discussed within this proposed amendment.

Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The intent of this criterion is to factor risk insights and operating experience into the TS limiting conditions for operation. Risk is significantly reduced with the reactor in the permanently defueled condition.

10 CFR 50.36(c)(5) Administrative Controls states, in part:

Administrative controls 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.

The particular administrative controls to be included in the TS generally are requirements the NRC deems necessary to support the safe operation of a facility that are not already covered by other regulations. These requirements are predominately specified in support of an operating plant. Once DBNPS is in a permanently shutdown and defueled condition, certain administrative controls described in the TS will no longer apply and will be deleted or modified.

10 CFR 50.36(c)(6) Decommissioning states:

This paragraph applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1) and to non-power reactor facilities which are not authorized to operate. Technical specifications 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.

As previously noted, FENOC provided formal notification to the NRC by letter dated April 25, 2018 (Reference 1) of FENOC's decision to permanently cease power operations at DBNPS by May 31, 2020. After docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel. As a result, DBNPS will be authorized to only possess special nuclear material. This proposed amendment deletes DBNPS TS that are no longer applicable to a permanently defueled facility while modifying some of the remaining TS to correspond to the permanently shut down condition.

### **3.1.3 10 CFR 50.48 Fire Protection**

10 CFR 50.48(f) establishes the requirement for maintaining a fire protection program once licensees have submitted the certifications required under 10 CFR Part 50.82(a)(1):

(f) Licensees that have submitted the certifications required under § 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). A fire protection program that complies with NFPA 805 shall be deemed to be acceptable for complying with the requirements of this paragraph.

As previously noted, FENOC provided formal notification to the NRC by letter dated April 25, 2018 (Reference 1) of FENOC's decision to permanently cease power operations at DBNPS by May 31, 2020. Since the initial certification has been submitted pursuant to 10 CFR 50.82(a)(1)(i) (Reference 1) and once the final certification required by 10 CFR 50.82(a)(1)(ii) has been submitted, the requirements of 10 CFR 50.48(f) will be in full effect.

### **3.1.4 DBNPS UFSAR Design Basis Accidents**

Chapter 15 of the DBNPS UFSAR describes the DBA and transient scenarios applicable to DBNPS. With the termination of reactor operations at DBNPS and the permanent removal of fuel from the reactor as certified in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the majority of the DBA scenarios postulated in the UFSAR will no longer be possible. During decommissioning the irradiated fuel will be stored in the SFP, or in the DFSF, until it is shipped off site in accordance with the schedules to be provided in the PSDAR and the Spent Fuel

Management Plan. The RCS, steam system, and turbine generator are no longer in operation and have no function related to the safe storage and management of the spent nuclear fuel. Table 2.1 lists the design basis accidents applicable to DBNPS after it is permanently defueled.

### **3.1.5 10 CFR 50.51, Continuation of License**

10 CFR 50.51(b) 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.

### **3.1.6 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Reactors**

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

### **3.1.7 10 CFR 50.62 Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants**

10 CFR 50.62(a) 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."

## **3.2 No Significant Hazards Consideration Analysis**

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," FirstEnergy Nuclear Operating Company (FENOC), proposes an amendment to the Renewed Facility Operating License (RFOL) and Appendix A, technical specifications (TS), of RFOL No. NPF-3 for Davis-Besse Nuclear Power

Station, Unit No. 1 (DBNPS). The proposed license amendment request (LAR) would revise the RFOL and the associated TS to the permanently defueled technical specifications (PDTs) consistent with the permanent cessation of power operation and permanent defueling of the reactor.

The proposed changes would revise and remove certain requirements contained within the RFOL and TS and remove requirements that would no longer be applicable once it has been certified that all fuel has permanently been removed from the reactor. Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed, the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2).

FENOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No.

The proposed changes would not take effect until DBNPS has certified to the NRC that it has permanently ceased operation and entered a permanently defueled condition. Because the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel with the certifications required by 10 CFR Part 50.82(a)(1) submitted, as specified in 10 CFR Part 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible.

The remaining UFSAR Chapter 15 postulated design basis accident (DBA) events that could potentially occur at a permanently defueled facility would be a fuel handling accident (FHA) in the spent fuel pool (SFP), the waste gas decay tank rupture (WGDTR), and external causes. The FHA analyses for DBNPS shows that, following 95 days of decay time after reactor shutdown and provided the SFP water level requirements of TS LCO 3.7.14 are met, the dose consequences are acceptable without relying on structures, systems, and components (SSCs) to remain functional for accident mitigation during and following the event other than the passive SFP structure. The remaining DBAs that support the permanently shutdown and defueled condition do not rely on any active safety systems for mitigation.

The probability of occurrence of previously evaluated accidents is not increased, since safe storage and handling of fuel will be the only operations performed, 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.

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 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 DBNPS 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.

Therefore, the proposed change 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 delete or modify certain DBNPS RFOL, TS, and current licensing bases (CLB) have no impact on facility SSCs affecting the safe storage of spent irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of spent irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor, or only to the prevention, diagnosis, or mitigation of reactor related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shutdown and defueled.

The proposed modification or deletion of requirements of the DBNPS RFOL, TS, and CLB do not affect systems credited in the accident analysis for the remaining credible DBAs at DBNPS. The proposed RFOL and PDTs will continue to require proper control and monitoring of safety significant parameters and activities. The TS regarding SFP water level and spent fuel storage is retained to preserve the current requirements for safe storage of irradiated fuel. 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, spent fuel racks, SFP integrity, and SFP water level). Since extended operation in a defueled condition and safe fuel handling will be the only operation 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 change 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.

The proposed changes are to delete or modify certain RFOL, TS, and CLB once the DBNPS facility has been permanently shutdown and defueled. Because the 10 CFR Part 50 license for DBNPS will no longer authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel, the occurrence of postulated accidents associated with reactor operation is no longer credible. The remaining postulated DBA events that could potentially occur at a permanently defueled facility would be a FHA, WGDTR, and external causes. The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses.

The proposed changes are limited to those portions of the RFOL, TS, and CLB that are not related to the safe storage of irradiated fuel. The requirements that are proposed to be revised or deleted from the RFOL, TS, and CLB are not credited in the updated applicable accident analysis for the remaining applicable postulated accidents, and as such, do not contribute to the margin of safety associated with the accident analysis. Postulated design basis accidents involving the reactor will no longer be possible because the reactor will be permanently shutdown and defueled, and DBNPS will no longer be authorized to operate the reactor.

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

Based on the above, FENOC concludes that the proposed amendment does not involve a 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 Conclusions**

In 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.

### **4.0 ENVIRONMENTAL CONSIDERATION**

FENOC has evaluated this license amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

### **5.0 REFERENCES**

1. Letter from Donald A. Moul (FES) to NRC Document Control Desk, "Certification of Permanent Cessation of Power Operations for Beaver Valley Power Station, Unit Nos. 1 and 2, Davis-Besse Nuclear Power Station, Unit No. 1, and Perry Nuclear Power Plant, Unit No. 1," dated April 25, 2018 (ADAMS Accession No. ML18115A007).
2. Crystal River Unit 3 Nuclear Generating Plant – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC No MF3089), dated September 4, 2015 (ADAMS Accession No. ML15224B286).
3. 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) dated July 17, 2015 (ADAMS Accession No. ML15139A390).

4. Kewaunee Power Station – Issuance of Amendment for Permanently Shutdown and Defueled Technical Specifications and Certain License Conditions (TAC No. MF1952), dated February 13, 2015 (ADAMS Accession No. ML14237A045).
5. 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), dated March 6, 2018 (ADAMS Accession No. ML18010A087).
6. Vermont Yankee Nuclear Power Station – Issuance of Amendment for Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition (CAC No. MF3714), dated October 7, 2015 (ADAMS Accession No. ML15117A551).
7. NRC Safety Evaluation Report for Millstone Power Station Unit 1 in License Amendment 106 to DPR-21, dated November 9, 1999 (ADAMS Accession No. ML993330283 and ML993330269).
8. NRC Safety Evaluation for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively (License Nos. DPR-39 and DPR-48)), dated December 30, 1999 (ADAMS Accession Nos. ML003672704 and ML003672696).
9. Letter from Brian D. Boles (FENOC) to NRC Document Control Desk “Notification of Completion of License Renewal Commitments,” dated November 18, 2016 (ADAMS Accession No. ML16327A066).
10. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Rev. 0 (Released March 1972, Withdrawn December 2016).
11. NRC Letter to Entergy Nuclear Operations, Inc., Vermont Yankee Nuclear Power Station – Request for Exemption from the Requirements of 10 CFR 50.54(m) (TAC No. MF2990), dated June 18, 2014 (ML14147A216).
12. Oyster Creek Nuclear Generating Station – Issuance of Amendment Regarding Changes to the Administrative Controls Section of the Technical Specifications (CAC No. MF8108), dated March 7, 2017 (ADAMS Accession No. ML16235A413).

13. Pilgrim Nuclear Power Station – Issuance of Amendment No. 246, Revise Administrative Controls Section of Technical Specifications to Change Staffing and Training Requirements for Permanently Defueled Condition (CAC No. MF9304), dated July 10, 2017 (ADAMS Accession No. ML17066A130).
14. Vermont Yankee – Issuance of Amendment No. 260, Revise and Remove Certain Requirements from Technical Specification Section 6.0, “Administrative Controls,” No Longer Applicable for its Permanently Defueled Condition (TAC No. MF2991), dated December 22, 2014 (ADAMS Accession No. ML14217A072).
15. FENOC Letter to NRC, “Request for Approval of Certified Fuel Handler Training Program,” dated August 15, 2018 (Accession No. ML18227A019).
16. Davis-Besse Nuclear Power Station, Unit No. 1 – Issuance of Amendment for the Conversion to the Improved Technical Specifications with Beyond Scope Issues (TAC Nos. MD6319-MD6319MD6322, MD6324-MD6333, MD6398-MD6403, MD6644-MD6649, and MD6684), dated November 20, 2008 (ADAMS Accession No. ML082900600).
17. Letter from M. Bezilla to NRC Document Control Desk, “License Amendment Request – Proposed Changes to Technical Specifications Sections 1.1, “Definitions,” and 5.0, “Administrative Controls,” for Permanently Defueled Condition,” dated October 22, 2018 (ADAMS Accession No. ML18295A289).
18. Letter from P. Harden to NRC Document Control Desk, “Supplemental Information Regarding a Pending Administrative License Amendment Request to Reflect a Change in the Entity Providing a \$400 Million Support Agreement,” dated August 23, 2018 (ADAMS Accession No. ML18235A194).

## Attachment 1

### License and Technical Specification Page Markups

(86 pages follow)

Technical Specifications that are deleted in their entirety are identified as such in the Technical Specification Table of Contents, however, the associated deletions are not included in this attachment. The remaining Technical Specifications are intentionally not re-numbered.



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

FIRSTENERGY NUCLEAR OPERATING COMPANY

AND

FIRSTENERGY NUCLEAR GENERATION, LLC

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-3

1. The Nuclear Regulatory Commission (the Commission) having found that:
  - A. The application for renewed license filed by FirstEnergy Nuclear Operating Company (FENOC)<sup>1</sup>, acting on its own behalf and as agent for FirstEnergy Nuclear Generation, LLC (licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;  
Deleted per Amendment No. ###.
  - B. ~~Construction of the Davis-Besse Nuclear Power Station, Unit No. 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-80 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;~~  
be maintained
  - C. The facility will ~~operate~~ in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

---

<sup>1</sup> FENOC is authorized to act as agent for FirstEnergy Nuclear Generation, LLC, and has exclusive responsibility and control over the physical construction, ~~operation~~, and maintenance of the facility.



- 1.D. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;
- E. The FirstEnergy Nuclear Operating Company is technically qualified and the licensees are financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;
- F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;
- G. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;
- H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility Operating License No. NPF-3 subject to the conditions for protection of the environment set forth herein is in accordance with 10 CFR Part 51 (formerly Appendix D to 10 CFR Part 50), of the Commission's regulations and all applicable requirements have been satisfied;
- ~~I. 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 Part 30, 40, and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31; and~~ facility maintenance
- J. 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 the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.
2. Renewed Facility Operating License No. NPF-3 is hereby issued to FirstEnergy Nuclear Operating Company (FENOC), and FirstEnergy Nuclear Generation, LLC to read as follows:
- A. This renewed license applies to the Davis-Besse Nuclear Power Station, Unit No. 1, a pressurized water nuclear reactor and associated equipment

permanently  
defueled

(the facility), owned by FirstEnergy Nuclear Generation, LLC. The facility is located on the south-western shore of Lake Erie in Ottawa County, Ohio, approximately 21 miles east of Toledo, Ohio, and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 14 through 44) and the Environmental Report as supplemented and amended (Supplements 1 through 2).



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

- (1) FENOC, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, ~~use, and operate~~ the facility; and use
- (2) FirstEnergy Nuclear Generation, LLC, to possess the facility at the designated location in Ottawa County, Ohio in accordance with the procedures and limitations set forth in this renewed license;
- (3) FENOC, 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 Safety Analysis Report, as supplemented and amended; as required for nuclear fuel storage;
- (4) FENOC, 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~~; that was used
- (5) FENOC, 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; and
- (6) FENOC, 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~~.

or

that were

and to possess any byproduct, source and special nuclear material as sealed neutron sources previously used for reactor startup and reactor instrumentation; and fission detectors;



- 2.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 or incorporated below:

(1) Maximum Power Level

Deleted per Amendment No. ###.

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

(2) Technical Specifications

###

The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby incorporated in the renewed license. FENOC shall operate the facility in accordance with the Technical Specifications.

maintain

Permanently Defueled

(3) Additional Conditions

Deleted per Amendment No. ###.

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

- (a) ~~FENOC shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation.~~
- (b) ~~Deleted per Amendment 6~~
- (c) ~~Deleted per Amendment 5~~

###

- 2.C(3)(d)** ~~Prior to operation beyond 32 Effective Full Power Years, FENOG shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.~~
- (e)** ~~Deleted per Amendment 33~~
  - (f)** ~~Deleted per Amendment 33~~
  - (g)** ~~Deleted per Amendment 33~~
  - (h)** ~~Deleted per Amendment 24~~
  - (i)** ~~Deleted per Amendment 11~~
  - (j)** ~~Revised per Amendment 3~~  
~~Deleted per Amendment 28~~
  - (k)** ~~Within 60 days of startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall complete tests and obtain test results as required by the Commission to verify that faults on non-Class 1E circuits would not propagate to the Class 1E circuits in the Reactor Protection System and the Engineered Safety Features Actuation System.~~
  - (l)** ~~Revised per Amendment 7~~  
~~Deleted per Amendment 15~~
  - (m)** ~~Deleted per Amendment 7~~
  - (n)** ~~Deleted per Amendment 10~~
  - (o)** ~~Deleted per Amendment 2~~
  - (p)** ~~Deleted per Amendment 29~~
  - (q)** ~~Deleted per Amendment 7~~
  - (r)** ~~Deleted per Amendment 30~~
  - (s)** ~~Toledo Edison Company shall be exempted from the requirements of Technical Specification 3/4.7.8.1 for the two (2) Americium-Beryllium-Copper startup sources to be installed or already installed for use during the first refueling cycle until such time as the sources are replaced.~~
  - (t)** ~~Added per Amendment 83~~  
~~Deleted per Amendment 122~~



Deleted per Amendment No. ###.

2.C(4) Fire Protection

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

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

(5) Deleted per Amendment No. 279.

(6) Antitrust Conditions

Deleted per Amendment No. ###.

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



2.C(7) Steam Generator Tube Circumferential Crack Report

~~Following each inservice inspection of steam generator tubes, the NRC shall be notified by FENOC of the following prior to returning the steam generators to service:~~

- ~~a. Indications of circumferential cracking inboard of the roll repair.~~
- ~~b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube to tubesheet seal weld if no reroll is present.~~
- ~~c. Determination of the best estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube to tubesheet rolls, tube to tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.~~

~~FENOC shall demonstrate by evaluation that the primary to secondary leakage following a LBLOCA, if any, as described in Appendix A to topic Report BAW-2374, July 2000, continues to be acceptable, based on the as-found condition of the steam generators. For the purpose of this evaluation, acceptable means that a best estimate of the leakage expected in the event of a LBLOCA would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR Part 100 limits). This is required to demonstrate that adequate margin and defense in depth continue to be maintained. A written summary of this evaluation shall be provided to the NRC within three months following completion of the steam generator tube inservice inspection.~~

2.C(8) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions that include the follow key area:

- (a) Fire fighting response strategy with the following elements:
  - 1. Predefined 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



Deleted per Amendment No. ###.

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

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

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

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

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

Deleted per Amendment No. ###.

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

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

Deleted per Amendment No. ###.

(11) License Renewal License Conditions

- (a) The information in the Updated Final Safety Analysis Report (UFSAR) supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by the Commitments applicable to Davis-Besse Nuclear Power Station, Unit No. 1, in Appendix A of the "Supplemental Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station" (SER) dated August 2015, is collectively the "License Renewal UFSAR Supplement." The License Renewal 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 and activities applicable to Davis-Besse Nuclear Power Station, Unit No. 1, described in the License Renewal UFSAR Supplement provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.



(b) ~~This License Renewal UFSAR Supplement, as revised per License Condition 11(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation.~~

- ~~1. The licensee shall implement these new programs and enhancements to existing programs no later than October 22, 2016.~~
- ~~2. The licensee shall complete these activities as noted in the Commitments applicable to Davis-Besse Nuclear Power Station, Unit No. 1, in the License Renewal UFSAR Supplement no later than October 22, 2016 or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.~~
- ~~3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.~~

(e) ~~This license condition requires testing of surveillance capsules for the period of extended operation to meet the test procedures and reporting requirements of American Society of Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. All pulled capsules shall be properly maintained for testing, and any changes to storage requirements must be approved by the NRC. All pulled and tested capsules, unless discarded before August 31, 2000, shall be placed in storage to be saved for possible future reconstitution and use.~~

- 2.D. FENOC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Davis-Besse Nuclear Power Station Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan Revision 4," submitted by letter dated May 18, 2006.

FENOC 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 FENOC CSP was approved by License Amendment No. 283 and is amended by License Amendment No. 290.

Deleted per Amendment No. ###.

- E. ~~This license is subject to the following antitrust conditions:~~

#### Definitions

~~Entity shall mean any electric generation and/or distribution system or municipality or cooperative with a statutory right or privilege to engage in either of these functions.~~

~~Wheeling shall mean transportation of electricity by a utility over its lines for another utility, including the receipt from and delivery to another system of like amounts but not necessarily the same energy. Federal Power Commission, The 1970 National Power Survey, Part 1, p. 1-24-8.~~

#### License Conditions Approved By the Atomic Safety and Licensing Appeal Board\*

- (1) ~~Applicants shall not condition the sale or exchange of wholesale power or coordination services upon the condition that any other entity:~~
- (a) ~~enter into any agreement or understanding restricting the use of or alienation of such energy or services to any customers or territories;~~

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\* ~~"Applicants" as used by the Appeal Board refers to the Toledo Edison Company, Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company and Pennsylvania Power Company although none of these entities are currently Licensees for this facility.~~

- 2.E(1)(b) enter into any agreement or understanding requiring the receiving entity to give up any other power supply alternatives or to deny itself any market opportunities;
- (e) withdraw any petition to intervene or forego participation in any proceeding before the Nuclear Regulatory Commission or refrain from instigating or prosecuting any antitrust action in any other forum.
- (2) Applicants, and each of them, shall offer interconnections upon reasonable terms and conditions at the request of any other electric entity(ies) in the CCCT, such interconnections to be available (with due regard for any necessary and applicable safety procedures) for operation in a closed switch synchronous operating mode if requested by the interconnecting entity(ies). Ownership of transmission lines and switching stations associated with such interconnection shall remain in the hands of the party funding the interconnection subject, however, to any necessary safety procedures relating to disconnection facilities at the point of power delivery. Such limitations on ownership shall be the least necessary to achieve reasonable safety practices and shall not serve to deprive purchasing entities of a means to effect additional power supply options.
- (3) Applicants shall engage in wheeling for and at the request of other entities in the CCCT:
- (a) of electric energy from delivery points of Applicants to the entity(ies); and;
- (b) of power generated by or available to the other entity, as a result of its ownership or entitlements\* in generating facilities, to delivery points of applicants designated by the other entity.
- (c) The Cleveland Electric Illuminating Company shall file with the FERC, within ten (10) days of the Order of the Director of Nuclear Reactor Regulation dated May 13, 1980, an amendment to its January 27, 1978 Transmission Service Schedule, FERC Docket ER78-194, in accordance with Appendix A to that Order and in conformity with the applicable filing requirements of the Federal Energy Regulatory Commission.

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\* "Entitlement" includes but is not limited to power made available to an entity pursuant to an exchange agreement.

2.E(3)(cont.)

~~Such wheeling services shall be available with respect to any unused capacity on the transmission lines of Applicants, the use of which will not jeopardize Applicants' system. In the event Applicants must reduce wheeling services to other entities due to lack of capacity, such reduction shall not be effected until reductions of at least 5% have been made in transmission capacity allocations to other Applicants in these proceedings and thereafter shall be made in proportion to reductions\* imposed upon other Applicants to this proceeding.~~

~~Applicants shall make reasonable provisions for disclosed transmission requirements of other entities in the CCCT in planning future transmission either individually or within the CAPCO grouping. By "disclosed" is meant the giving of reasonable advance notification of future requirements by entities utilizing wheeling services to be made available by Applicants.~~

- ~~(4)(a) Applicants shall make available membership in CAPCO to any entity in the CCCT with a system capability of 10 MW or greater;~~
- ~~(b) A group of entities with an aggregate system capability of 10 MW or greater may obtain a single membership in CAPCO on a collective basis.\*\*~~

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\* ~~The objective of this requirement is to prevent preemption of unused capacity on the lines of one Applicant by other Applicants or by entities the transmitting Applicant deems noncompetitive. Competitive entities are to be allowed opportunity to develop bulk power services options even if this results in reallocation of CAPCO transmission channels. This relief is required in order to avoid prolongation of the effects of Applicants' illegally sustained dominance.~~

\*\* ~~E.g., Wholesale Customer of Ohio Edison (WCOE).~~

- 2.E(4)(e) ~~Entities applying for membership in CAPCO pursuant to License Condition 4 shall become members subject to the terms and conditions of the CAPCO Memorandum of Understanding of September 14, 1967, and its implementing agreements, except that new members may elect to participate on an equal percentage of reserve basis rather than a P/N allocation formula for a period of twelve years from date of entrance.\* Following the twelfth year of entrance, new members shall be expected to adhere to such allocation methods as are then employed by CAPCO (subject to equal opportunity for waiver or special consideration granted to original CAPCO members which then are in effect).~~
- (d) ~~New members joining CAPCO pursuant to this provision of relief shall not be entitled to exercise voting rights until such time as the system capability of the joining member equals or exceeds the system capability of the smallest member of CAPCO which enjoys voting rights.\*\*~~

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\* ~~The selection of the 12 year period reflects our determination that an adjustment period is necessary since the P/N formula has a recognized effect of discriminating against small systems and forcing them to forego economies of scale in generation in order to avoid carrying excessive levels of reserves. We also found that P/N is not entirely irrational as a method of reserve allocation. We have observed that Applicants themselves provided adjustment periods and waivers to integrate certain Applicants into the CAPCO reserve requirement program. The 12 year period should permit new entrants to avoid initial discrimination but to accommodate and adjust to the CAPCO system over some reasonable period of time. Presumably new entrants will be acquiring ownership shares and entitlements during the 12 year period so that adverse consequences of applying the P/N formula will be mitigated.~~

\*\* ~~Our objective is to prevent impediments to the operation and development of an areawide power pool through the inability of lesser entities to respond timely or to make necessary planning commitments. While we grant new member entities the opportunity to participate in CAPCO it is not our intent to relieve joining entities of responsibilities and obligations necessary to the successful operation of the pool. For those smaller entities which do not wish to assume the broad range of obligations associated with CAPCO membership we have provided for access to bulk power service options which will further their ability to survive and offer competition in the CCGT.~~



- 2.E(5) ~~Applicants shall sell maintenance power to requesting entities in the CCCT upon terms and conditions no less favorable than those Applicants make available: (1) to each other either pursuant to the CAPCO agreements or pursuant to bilateral contract; or (2) to non-Applicant entities outside the CCCT.~~
- (6) ~~Applicants shall sell emergency power to requesting entities in the CCCT upon terms and conditions no less favorable than those Applicants make available: (1) to each other pursuant to the CAPCO agreements or pursuant to bilateral contract; or (2) to non-Applicant entities outside the CCCT.~~
- (7) ~~Applicants shall sell economy energy to requesting entities in the CCCT, when available, on terms and conditions no less favorable than those available: (1) to each other either pursuant to the CAPCO agreements or pursuant to bilateral contract; or (2) to non-Applicant entities outside the CCCT.~~
- (8) ~~Applicants shall share reserves with any interconnected generation entity in the CCCT upon request. The requesting entity shall have the option of sharing reserves on an equal percentage basis or by use of the CAPCO P/N allocation formula or on any other mutually agreeable basis.~~
- (9) (a) ~~Applicants shall make available to entities in the CCCT access to the Davis-Besse 1, 2 and 3 and the Perry 1 and 2 nuclear units and any other nuclear units for which Applicants or any of them, shall apply for a construction permit or operating license during the next 25 years. Such access, at the option of the requesting entity, shall be on an ownership share, or unit participation or contractual pre-purchase of power basis.\*~~
- ~~Each requesting entity (or collective group of entities) may obtain up to 10% of the capacity of the Davis-Besse and Perry Units and 20% of future units (subject to the 25-year limitation) except that once any entity or entities have contracted for allocations totaling 10% or 20%, respectively, no further participation in any given unit need be offered.~~

---

\* ~~Requesting entities' election as to the type of access may be affected by provisions of state law relating to dual ownership of generation facilities by municipalities and investor-owned utilities. Such laws may change during the period of applicability of these conditions. Accordingly, we allow requesting entities to be guided by relevant legal and financial considerations (including Commission regulations on nuclear power plant ownership) in fashioning their requests.~~

**2.E(9)(b)** ~~Commitments for the Davis-Besse and Perry Units must be made by requesting entities within two years after this decision becomes final. Commitments for future units must be made within two years after a construction permit application is filed with respect to such a unit (subject to the 25-year limitation) or within two years after the receipt by a requesting entity of detailed written notice of Applicants' plans to construct the unit, whichever is earlier; provided, however, that the time for making the commitment shall not expire until at least three months after the filing of the application for a construction permit. Where an Applicant seeks to operate a nuclear plant with respect to which it did not have an interest at the time of the filing of the application for the construction permit, the time periods for commitments shall be the same except that reference should be to the operating license, not the construction permit.~~

**(10)** ~~Applicants shall sell wholesale power to any requesting entity in the GGCT, in amounts needed to meet all or part of such entity's requirements. The choice as to whether the agreement should cover all or part of the entity's requirements should be made by the entity, not the Applicant or Applicants.~~

**(11)** ~~These conditions are intended as minimum conditions and do not preclude Applicants from offering additional wholesale power or coordination services to entities within or without the GGCT. However, Applicants shall not deny wholesale power or coordination services required by these conditions to non-Applicant entities in the GGCT based upon prior commitments arrived at in the CAPCO Memorandum of Understanding or implementing agreements. Such denial shall be regarded as inconsistent with the purpose and intent of these conditions.~~

~~The above conditions are to be implemented in a manner consistent with the provisions of the Federal Power Act and all rates, charges or practices in connection therewith are to be subject to the approval of regulatory agencies having jurisdiction over them.~~

2.F. This renewed license is subject to the following additional conditions for the protection of the environment:

- (1) FENOC shall ~~operate~~ <sup>maintain</sup> Davis-Besse Unit No. 1 within applicable Federal and State air and water quality standards.
- (2) Before engaging in an ~~operational~~ activity not evaluated by the Commission, FENOC will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement, FENOC shall provide a written evaluation of such activities and obtain prior approval of the Director, Office of Nuclear Reactor Regulation for the activities.



Deleted per Amendment No. ###.

This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

I. Handling of irradiated fuel that has occupied part of a critical reactor core within the previous 95 days is not permitted.

- G. ~~In accordance with the requirement imposed by the October 8, 1976, order of the United States Court of Appeals for the District of Columbia Circuit in Natural Resources Defense Council v. Nuclear Regulatory Commission, No. 74-1385 and 74-1586, that the Nuclear Regulatory Commission "shall make any licenses granted between July 21, 1976 and such time when the mandate is issued subject to the outcome of such proceedings herein," this license shall be subject to the outcome of such proceedings.~~
- H. ~~This renewed license is effective as of the date of issuance and shall expire at midnight April 22, 2037.~~

3. Based on the Commission's Order dated December 16, 2005 and conforming Amendment No. 270 dated December 16, 2005 regarding the direct transfer of the license from the ~~Cleveland Electric Illuminating Company (Cleveland Electric)~~ and the ~~Toledo Edison Company (Toledo Edison)~~ to ~~FirstEnergy Nuclear Generation Corp. (FENGenCo)\*~~, ~~FirstEnergy Nuclear Operating Company~~ and ~~FENGenCo~~ shall comply with the following conditions noted below:

- A. ~~On the closing date of the transfers to FENGenCo\* of their interests in Davis-Besse, Cleveland Electric and Toledo Edison shall transfer to FENGenCo\* all of each transferor's respective accumulated decommissioning funds for Davis-Besse and tender to FENGenCo\* additional amounts equal to remaining funds expected to be collected in 2005, as represented in the application dated June 1, 2005, but not yet collected by the time of closing. All of the funds shall be deposited in a separate external trust fund for the reactor in the same amount as received with respect to the unit to be segregated from other assets of FENGenCo\* and outside its administrative control, as required by NRC regulations, and FENGenCo\* shall take all necessary steps to ensure that this external trust fund is maintained in accordance with the requirements of the order approving the transfer of the license and consistent with the safety evaluation supporting the order and in accordance with the requirements of 10 CFR Section 50.75, "Reporting and recordkeeping for decommissioning planning."~~

FirstEnergy Nuclear Generation LLC

Order

the decommissioning

Order

dated December 16, 2005

FirstEnergy Nuclear Generation LLC

\* ~~FirstEnergy Nuclear Generation Corp. (FENGenCo)\* has been renamed FirstEnergy Nuclear Generation, LLC.~~



Deleted per Amendment No. ###.



- B. ~~The Support Agreement described in the application dated June 1, 2005 (up to \$400 million), shall be effective consistent with the representations contained in the application. FENGenCo\* shall take no action to cause FirstEnergy, or its successors and assigns, to void, cancel, or modify the Support Agreement without the prior written consent of the NRC staff. FENGenCo\* shall inform the Director of the Office of Nuclear Reactor Regulation, in writing, no later than ten days after any funds are provided to FENGenCo\* by FirstEnergy under either Support Agreement.~~

FOR THE NUCLEAR REGULATORY COMMISSION

William M. Dean, Director  
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A - Technical Specifications
2. ~~Preoperational Tests, Startup Tests and Other Items Which Must Be Completed Prior to Proceeding to Succeeding Operational Modes~~

Deleted per Amendment No. ###.

Date of Issuance: December 8, 2015

\* ~~FirstEnergy Nuclear Generation Corp. (FENGenCo)~~ has been renamed ~~FirstEnergy Nuclear Generation, LLC.~~



## ATTACHMENT 2 TO LICENSE NPF-3

### Preoperational Test, Startup Tests, and Other Items Which Must be Completed Prior to Proceeding to Succeeding Operational Modes

~~This attachment identifies certain preoperational tests, startup tests, and other items which must be completed to the Commission's satisfaction prior to proceeding to certain specified Operational Modes. Toledo Edison Company shall not proceed beyond the authorized Operational Modes without prior written authorization from the Commission.~~

- A. ~~Toledo Edison Company may at the license issue date proceed directly to Operational Mode 6 (initial fuel loading), and may subsequently proceed to Operational Mode 5 (cold shutdown), except as noted below.~~
- B. ~~The following items must be completed prior to proceeding to Operational Mode 5 (cold shutdown):~~

1. ~~Approval is required of the fifteen listed surveillance procedures~~

|                      |   |
|----------------------|---|
| <del>ST5030.02</del> | <del>RPS Monthly Check</del>                                      |
| <del>ST5030.00</del> | <del>RPS Response Time</del>                                      |
| <del>ST5031.14</del> | <del>RPS Response Time Calculation</del>                          |
| <del>ST5036.02</del> | <del>Remote Shutdown Monitor Instrument Channel Calibration</del> |
| <del>ST5036.03</del> | <del>Post Accident Instrument Channel Check</del>                 |
| <del>ST5036.04</del> | <del>Post Accident Instrument Channel Calibration</del>           |
| <del>ST5050.02</del> | <del>Core Flood System Isolation Valve Check</del>                |
| <del>ST5051.01</del> | <del>ECGS Subsystem Monthly Test</del>                            |
| <del>ST5061.02</del> | <del>Containment Local Leak Test</del>                            |
| <del>ST5062.01</del> | <del>Containment Spray System Monthly Test</del>                  |
| <del>ST5070.01</del> | <del>Main Steam Safety Valve Setpoint</del>                       |
| <del>ST5020.01</del> | <del>Axial Power Imbalance Manual Calibration</del>               |
| <del>ST5022.03</del> | <del>Quadrant Power Tilt</del>                                    |
| <del>ST5033.02</del> | <del>Incore Monitor System Recorder Calibration</del>             |
| <del>ST5042.03</del> | <del>Reactor Coolant Flow Rate Test</del>                         |

2. ~~System Interaction~~

~~The Toledo Edison Company's 5000 and 8000 Series EIRs (Engineering Inspection Reports) concerning upgrading of supports and installation of water shields on non-safety related systems such that their failure will not degrade or cause failure of a safety related system must be completed as stated below:~~

- a. ~~Upgrading of 20 electrical tray and conduit supports primarily located in the 4160 and 480 volt switch gear rooms and Intake Water Structure.~~
  - b. ~~Final inspection and approval by Toledo Edison Company Quality Control of 24 completed modifications and approval by Engineering.~~
  - c. ~~Final review and approval by Toledo Edison Company Engineering of 26 completed and inspected modifications.~~
  - d. ~~Completion of 24 structural items, primarily shielding devices from water sources.~~
- G. The following items must be completed prior to proceeding to Operational Mode 4 (hot shutdown):
- 1. High Pressure Injection Pump Modification  
  
The Toledo Edison Company must provide documentation to establish that the modification work for the pumps is in accordance with the FSAR and the specification requirements.
  - 2. HVAC Systems  
  
The reinspection activity and subsequent repair effort, relative to welds needed to resist seismic design forces, must be completed.
  - 3. Large Pipe Hangers and Anchors  
  
Corrective action relative to 76 large pipe hangers and seven anchors for safety related systems must be completed.
  - 4. Small Pipe Hangers and Anchors  
  
Corrective action relative to small piping system discrepancies must be completed.
  - 5. Valve Stem Locknuts  
  
Stem locknuts for 141 valves with limit torque operators within safety related systems must be verified as being "staked."
  - 6. Leak Tightness Test of Valve Enclosure  
  
Approval of periodic test procedure and completion of a leak tightness test of the enclosure installed around DH 11 and DH 12 valves in containment.

7. Systems Interaction

- a. ~~Upgrading of 20 electrical conduit supports primarily located in hallways and corridors.~~
- b. ~~Upgrading of 27 pipe supports.~~

D. ~~The following items must be completed prior to proceeding to Operational Mode 3 (hot standby):~~

1. ~~Reworked Valves~~

~~Five small valves within safety related systems must be hydrostatically tested and accepted to the requirements of the applicable code.~~

E. ~~The following items must be completed prior to proceeding to Operational Mode 2 (initial criticality):~~

1. ~~Modification to replace the four level switches in each steam generator inside containment with four level transmitters.~~

2. ~~Resolution of discrepancies for Preoperational Tests:~~

~~PT 232.01, Miscellaneous Radwaste System  
PT 230.01, Clean Liquid Radwaste~~

3. ~~Completion of Preoperational Tests:~~

~~PT 230.02, Degassifier  
PT 230.03, Boric Acid Evaporator  
PT 231.02, Miscellaneous Waste Evaporator~~

F. ~~The following items must be completed prior to proceeding to Operational Mode 1 (power operation):~~

1. ~~Emergency Planning Procedures~~

- a. ~~An isolation emergency plan implementing procedure to cope with weather conditions which require personnel to remain at the station for undetermined periods shall be developed. This procedure shall also address provisions for transportation of emergency personnel to the station when needed during these periods.~~
- b. ~~The following topics will be incorporated into the Emergency Plan Implementing Procedure:~~

- (1) ~~Evacuation of personnel to minimize exposure to hazard.~~
- (2) ~~Personnel accountability to assist the person in charge of emergency response actions to account for missing persons.~~
- (3) ~~Reentry into previously evacuated areas for the purposes of saving lives, search and rescue of missing and injured persons. Safety equipment to be worn depending on areas or conditions shall be addressed.~~

2. ~~Completion of Preoperational Tests Solid Waste Compactor, PT-200.02.~~

3. Electrical Firebarrier Testing

~~The Toledo Edison Company shall provide documentation of fire barrier testing to assure conformance of the fire barriers installed at the Davis-Besse I plant to ASTS E-110.~~

4. Boron Dilution Mode Tests

~~Complete flow tests in the hot leg drain mode and the pressurizer spray mode to verify minimum flow of 40 gallons per minute.~~

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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.

| <u>Term</u>                                 | <u>Definition</u>   |
|---|---|
| ACTIONS                                     | ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.   |
| <del>ALLOWABLE THERMAL POWER</del>          | <del>ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.</del>   |
| <del>AXIAL POWER IMBALANCE</del>            | <del>AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP.</del>  |
| <del>AXIAL POWER SHAPING RODS (APSRs)</del> | <del>APSRs shall be control components used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.</del>   |
| <del>CHANNEL CALIBRATION</del>              | <del>A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.</del> |
| <del>CHANNEL CHECK</del>                    | <del>A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of</del>   |

## 1.1 Definitions

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### ~~CHANNEL CHECK (continued)~~

~~the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.~~

~~CHANNEL FUNCTIONAL TEST A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps.~~

### ~~CONTROL RODS~~

~~CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations.~~

### ~~CORE OPERATING LIMITS REPORT (COLR)~~

~~The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.~~

### ~~DOSE EQUIVALENT I-131~~

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

### ~~$\bar{E}$ —AVERAGE DISINTEGRATION ENERGY~~

~~$\bar{E}$  shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.~~

### ~~INSERVICE TESTING PROGRAM~~

~~The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).~~



## 1.1 Definitions

---

### LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal return flow), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal return flow) that is not identified LEAKAGE; and

c. Pressure Boundary LEAKAGE

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

### MODE

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

### NUCLEAR HEAT FLUX HOT CHANNEL FACTOR ( $F_Q$ )

$F_Q$  shall be the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions.

### NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ( $F_{\Delta H}^N$ )

$F_{\Delta H}^N$  shall be the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power.

## 1.1 Definitions

---

### ~~OPERABLE—OPERABILITY~~

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

### ~~PHYSICS TESTS~~

~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.~~

~~These tests are:~~

- ~~a. Described in Section 14, "Initial Tests and Operation," of the UFSAR;~~
- ~~b. Authorized under the provisions of 10 CFR 50.59; or~~
- ~~c. Otherwise approved by the Nuclear Regulatory Commission.~~

### ~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.~~

### ~~QUADRANT POWER TILT (QPT)~~

~~QPT shall be defined by the following equation and is expressed as a percentage of the Power in any Core Quadrant ( $P_{quad}$ ) to the Average Power of all Quadrants ( $P_{avg}$ ).~~

$$~~QPT = 100 [(P_{quad} / P_{avg}) - 1]~~$$

### ~~RATED THERMAL POWER (RTP)~~

~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2817 MWt.~~

## 1.1 Definitions

---

|   |  |
|---|--|
| REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME         | The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.  |
| SAFETY FEATURES ACTUATION SYSTEM (SFAS) RESPONSE TIME | The SFAS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its SFAS actuation setpoint at the channel sensor until the SFAS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.   |
| SHUTDOWN MARGIN (SDM)                                 | <p>SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:</p> <ul style="list-style-type: none"> <li>a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;</li> <li>b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and</li> <li>c. There is no change in APSR position.</li> </ul> |
| STAGGERED TEST BASIS                                  | A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, trains, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, trains, channels, or other designated components are tested during $n$ Surveillance Frequency intervals, where $n$ is the total number of systems, subsystems, trains, channels, or other designated components in the associated function.  |

## 1.1 Definitions

---

~~STEAM AND FEEDWATER  
RUPTURE CONTROL  
SYSTEM (SFRCS)  
RESPONSE TIME~~

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

~~THERMAL POWER~~

~~THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.~~

---

Table 1.1-1 (page 1 of 1)  
MODES

| MODE | TITLE                        | REACTIVITY<br>CONDITION<br>( $k_{eff}$ ) | % RATED<br>THERMAL<br>POWER <sup>(a)</sup> | AVERAGE<br>REACTOR COOLANT<br>TEMPERATURE<br>(°F) |
|------|------------------------------|--|--|---|
| 1    | Power Operation              | $\geq 0.99$                              | $> 5$                                      | NA  |
| 2    | Startup                      | $\geq 0.99$                              | $\leq 5$                                   | NA  |
| 3    | Hot Standby                  | $< 0.99$                                 | NA   | $\geq 280$  |
| 4    | Hot Shutdown <sup>(b)</sup>  | $< 0.99$                                 | NA   | $280 > T_{avg} > 200$                             |
| 5    | Cold Shutdown <sup>(b)</sup> | $< 0.99$                                 | NA   | $\leq 200$  |
| 6    | Refueling <sup>(c)</sup>     | 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    | <p>The purpose of this section is to explain the meaning of logical connectors.</p> <p>Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.</p>  |
| BACKGROUND | <p>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.</p> <p>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.</p> |
| EXAMPLES   | <p>The following examples illustrate the use of logical connectors.</p>  |

---

## 1.2 Logical Connectors

---

### EXAMPLES (continued)

#### EXAMPLE 1.2-1

##### ACTIONS

| CONDITION       | REQUIRED ACTION   | COMPLETION TIME |
|-----------------|---|-----------------|
| A. LCO not met. | A.1 Verify . . .<br><br><u>AND</u><br><br>A.2 Restore . . . |                 |

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

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

| CONDITION       | REQUIRED ACTION  | COMPLETION TIME |
|-----------------|--|-----------------|
| A. LCO not met. | <div>A.1 Trip . . .</div> <div>OR</div> <div>A.2.1 Verify . . .</div> <div>AND</div> <div>A.2.2.1 Reduce . . .</div> <div>OR</div> <div>A.2.2.2 Perform . . .</div> <div>OR</div> <div>A.3 Align . . .</div> |                 |

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

## 1.0 USE AND APPLICATION

### 1.3 Completion Times

|             |   |
|-------------|---|
| PURPOSE     | The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.   |
| BACKGROUND  | Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).  |
| DESCRIPTION | <p>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.</p> <p>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, "AG Sources — Operating," Required Action B.2, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied.</p> <p>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.</p> <p>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</p> |

handling and storage of  
nuclear fuel

facility

facility

### 1.3 Completion Times

#### DESCRIPTION (continued)

~~Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.~~

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

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

- ~~a. Must exist concurrent with the first inoperability; and~~
- ~~b. Must remain inoperable or not within limits after the first inoperability is resolved.~~

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

- ~~a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or~~
- ~~b. The stated Completion Time as measured from discovery of the subsequent inoperability.~~

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

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



### 1.3 Completion Times

#### EXAMPLES

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

#### EXAMPLE 1.3-1

#### ACTIONS

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

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

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

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

### 1.3 Completion Times

#### EXAMPLES (continued)

##### EXAMPLE 1.3.2

##### ACTIONS

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

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

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

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

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

### 1.3 Completion Times

#### EXAMPLES (continued)

##### EXAMPLE 1.3-3

##### ACTIONS

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

### 1.3 Completion Times

---

#### EXAMPLES (continued)

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

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

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

### 1.3 Completion Times

#### EXAMPLES (continued)

##### EXAMPLE 1.3-4

##### ACTIONS

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

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

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

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



### 1.3 Completion Times

#### EXAMPLES (continued)

##### EXAMPLE 1.3-5

##### ACTIONS

##### NOTE

Separate Condition entry is allowed for each inoperable valve.

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

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

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

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

### 1.3 Completion Times

#### EXAMPLES (continued)

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

#### EXAMPLE 1.3-6

##### ACTIONS

| CONDITION  | REQUIRED ACTION   | COMPLETION TIME  |
|--|---|------------------|
| A. One channel inoperable.                                 | A.1 Perform SR 3.x.x.x.                                   | Once per 8 hours |
|  | <u>OR</u><br>A.2 Reduce THERMAL POWER to $\leq 50\%$ RTP. | 8 hours          |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.   | 6 hours          |

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

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

### 1.3 Completion Times

#### EXAMPLES (continued)

##### EXAMPLE 1.3.7

##### ACTIONS

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

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

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

### 1.3 Completion Times

---

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

---

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## 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 | <p>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.</p> <p>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.</p> <p>Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.</p> <p>Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be preformed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.</p> <p>The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.</p> <p><del>Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:</del></p> |



## 1.4 Frequency

---

### DESCRIPTION (continued)

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

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

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### 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.~~



illustrates the type of frequency statement that appears in the Permanently Defueled Technical Specifications (PDTS).

## 1.4 Frequency

### EXAMPLES (continued)

#### EXAMPLE 1.4-1

##### SURVEILLANCE REQUIREMENTS

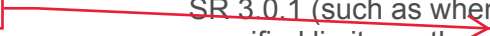
| SURVEILLANCE           | FREQUENCY |
|------------------------|-----------|
| Perform CHANNEL CHECK. | 12 hours  |

PDTS



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

facility



facility



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

## 1.4 Frequency

### EXAMPLES (continued)

#### EXAMPLE 1.4.2

#### SURVEILLANCE REQUIREMENTS

| <u>SURVEILLANCE</u>           | <u>FREQUENCY</u>  |
|-------------------------------|---|
| Verify flow is within limits. | Once within<br>12 hours after<br>≥ 25% RTP<br><br><u>AND</u><br><br>24 hours thereafter |

Example 1.4.2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4.1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

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

## 1.4 Frequency

### EXAMPLES (continued)

#### EXAMPLE 1.4-3

##### SURVEILLANCE REQUIREMENTS

| SURVEILLANCE  | FREQUENCY |
|---|-----------|
| <p><u>NOTE</u></p> <p>Not required to be performed until 12 hours after <math>\geq 25\%</math> RTP.</p> |           |
| 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  $\geq 25\%$  RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was  $< 25\%$  RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power  $\geq 25\%$  RTP.

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

## 1.4 Frequency

### EXAMPLES (continued)

#### EXAMPLE 1.4.4

##### SURVEILLANCE REQUIREMENTS

| SURVEILLANCE   | FREQUENCY |
|--|-----------|
| <p style="text-align: center;"><u>NOTE</u></p> <p>Only required to be met in MODE 1.</p> |           |
| Verify leakage rates are within limits.  | 24 hours  |

Example 1.4.4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4.1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency was not met), SR 3.0.4 would require satisfying the SR.

## 1.4 Frequency

### EXAMPLES (continued)

#### EXAMPLE 1.4-5

##### SURVEILLANCE REQUIREMENTS

| SURVEILLANCE   | FREQUENCY |
|--|-----------|
| <p style="text-align: center;"><u>NOTE</u></p> <p>Only required to be performed in MODE 1.</p> |           |
| Perform complete cycle of the valve.   | 7 days    |

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

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

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



## 1.4 Frequency

### EXAMPLES (continued)

#### EXAMPLE 1.4-6

#### SURVEILLANCE REQUIREMENTS

| SURVEILLANCE  | FREQUENCY |
|---|-----------|
| <p style="text-align: center;"><u>NOTE</u></p> <p>Not required to be met in MODE 3.</p> |           |
| Verify parameter is within limits.  | 24 hours  |

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

### 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

|           |   |
|-----------|---|
| LCO 3.0.1 | LCOs shall be met during the <del>MODES or other</del> specified conditions in the Applicability, except as provided in LCO 3.0.2, <del>LCO 3.0.7, and LCO 3.0.8.</del>   |
| LCO 3.0.2 | <p>Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met; <del>except as provided in LCO 3.0.5 and LCO 3.0.6.</del></p> <p><del>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.</del></p>   |
| LCO 3.0.3 | <p><del>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:</del></p> <ul style="list-style-type: none"> <li><del>a. MODE 3 within 7 hours;</del></li> <li><del>b. MODE 4 within 13 hours; and</del></li> <li><del>c. MODE 5 within 37 hours.</del></li> </ul> <p><del>Exceptions to this Specification are stated in the individual Specifications.</del></p> <p><del>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.</del></p> <p><del>LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.</del></p> |
| LCO 3.0.4 | <p><del>When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:</del></p> <ul style="list-style-type: none"> <li><del>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;</del></li> <li><del>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</del></li> </ul>  |

### 3.0 LCO Applicability

#### LCO 3.0.4 (continued)

- e. ~~When an allowance is stated in the individual value, parameter, or other Specification.~~

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

#### LCO 3.0.5

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

#### LCO 3.0.6

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

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

#### LCO 3.0.7

~~Test Exception LCOs 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.~~

### 3.0 LCO Applicability

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#### ~~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.~~

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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- 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.
- 
- SR 3.0.4      Entry into a ~~MODE or other~~ specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. ~~When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.~~

### 3.0 SR Applicability

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#### SR 3.0.4 (continued)

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

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### 3.7 PLANT SYSTEMS

#### 3.7.14 Spent Fuel Pool Water Level

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

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

#### ACTIONS

| CONDITION  | REQUIRED ACTION  | COMPLETION TIME |
|--|--|-----------------|
| A. Spent fuel pool water level not within limit. | <p>A.1 <del>NOTE</del><br/><del>LCO 3.0.3 is not applicable.</del></p> <hr/> <p>Suspend movement of irradiated fuel assemblies in spent fuel pool.</p> | Immediately     |

#### SURVEILLANCE REQUIREMENTS

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

### 3.7 PLANT SYSTEMS

#### 3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be  $\geq 630$  ppm.

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

#### ACTIONS

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

#### SURVEILLANCE REQUIREMENTS

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

### 3.7 PLANT SYSTEMS

#### 3.7.16 Spent Fuel Pool Storage

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

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

#### ACTIONS

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

#### SURVEILLANCE REQUIREMENTS

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

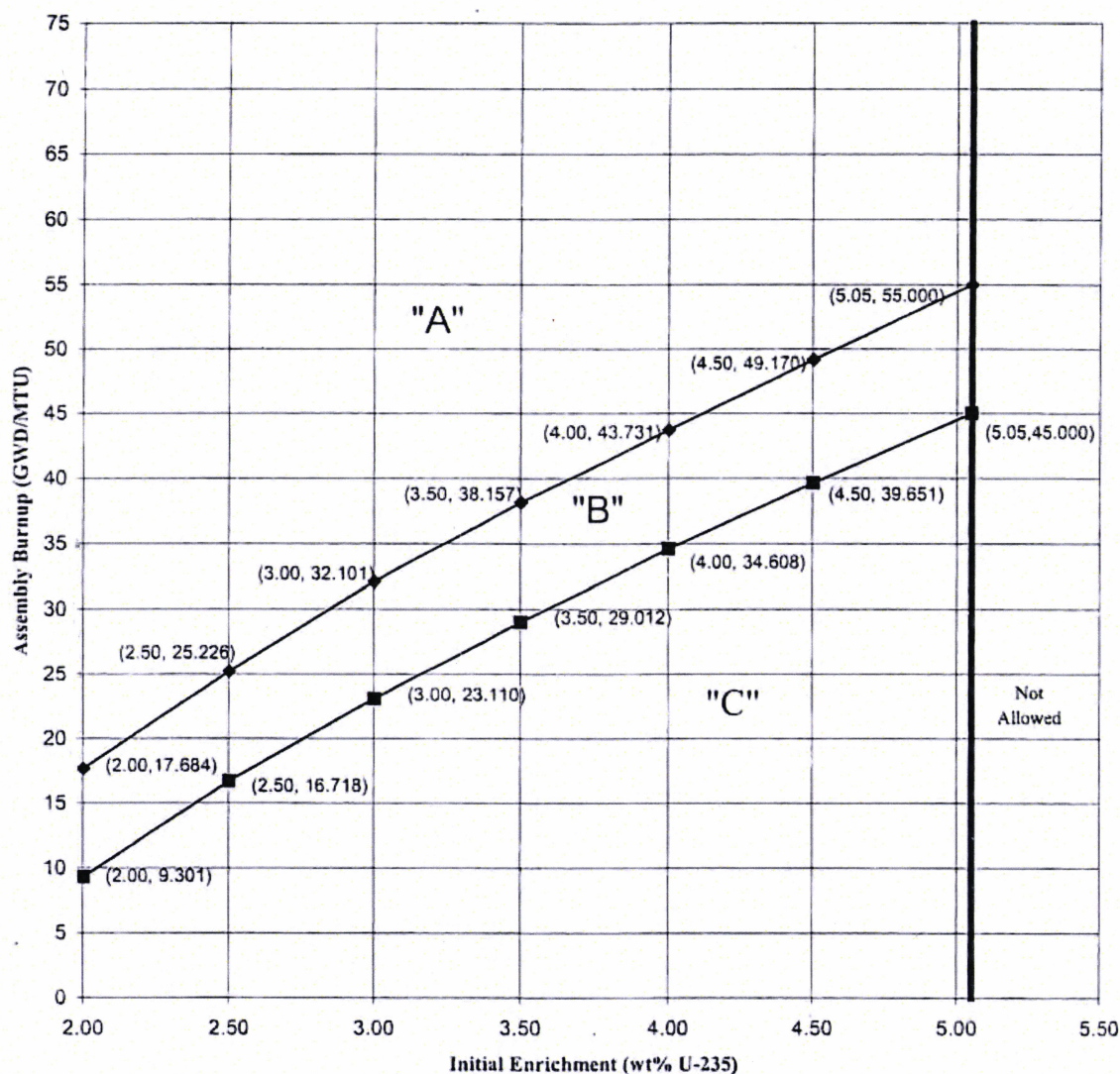


Figure 3.7.16-1 (page 1 of 1)  
Burnup versus Enrichment Curve for  
Spent Fuel Pool Storage Racks

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

## 4.0 DESIGN FEATURES

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### 4.1 Site Location

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

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### 4.2 Reactor Core

#### 4.2.1 Fuel Assemblies

~~The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy M5 or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO<sub>2</sub>) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.~~

#### 4.2.2 Control Rods

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

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### 4.3 Fuel Storage

#### 4.3.1 Criticality

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

- a.  $k_{eff} \leq 0.95$  if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;
- b. A nominal 9.22 inch center to center distance between fuel assemblies; and
- c. Fuel assemblies stored in the spent fuel storage racks in accordance with LCO 3.7.16.

## 4.0 DESIGN FEATURES

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### 4.3 Fuel Storage (continued)

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

- ~~a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;~~
- ~~b.  $k_{\text{eff}} \leq 0.95$  if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;~~
- ~~c.  $k_{\text{eff}} \leq 0.98$  when immersed in a hydrogenous mist, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and~~
- ~~d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.~~

#### 4.3.2 Drainage

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

#### 4.3.3 Capacity

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

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## 5.0 ADMINISTRATIVE CONTROLS

### 5.5 Programs and Manuals

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

#### 5.5.1 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.1 and Specification 5.6.2.
- c. Licensee initiated changes to the ODCM:
  1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
    - a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s); and
    - b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
  2. Shall become effective after the approval of the plant manager; and
  3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

## 5.5 Programs and Manuals

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### 5.5.2 Primary Coolant Sources Outside Containment

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

- ~~a. Preventive maintenance and periodic visual inspection requirements; and~~
- ~~b. Integrated leak test requirements for each system at least once per 24 months.~~

~~The provisions of SR 3.0.2 are applicable.~~

### 5.5.3 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 – 20.2402;
- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;
- e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

## 5.5 Programs and Manuals

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### 5.5.3 Radioactive Effluent Controls Program (continued)

- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;
- g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:
  - 1. For noble gases: a dose rate  $\leq 500$  mrem/yr to the whole body and a dose rate  $\leq 3000$  mrem/yr to the skin, and
  - 2. For iodine-131, iodine-133, tritium, and all other radionuclides in particulate form with half-lives  $> 8$  days: a dose rate  $\leq 1500$  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 to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
- i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives  $> 8$  days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and
- j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

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

### 5.5.4 Reactor Vessel Internals Vent Valves Program

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

- ~~a. Verify by visual inspection that the valve body and valve disc exhibit no abnormal degradation;~~
- ~~b. Verify the valve is not stuck in an open position; and~~
- ~~c. Verify by manual actuation that the valve is fully open when a force of  $\leq 400$  lbs is applied vertically upward.~~

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

## 5.5 Programs and Manuals

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### 5.5.5 Allowable Operating Transient Cycles Program

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

### 5.5.6 Reactor Coolant Pump Flywheel Inspection Program

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

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

### 5.5.7 Deleted

## 5.5 Programs and Manuals

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### 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 leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.~~

## 5.5 Programs and Manuals

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

- ~~3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."~~
- ~~e. 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.~~
- ~~4. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.~~



## 5.5 Programs and Manuals

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

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 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~~

## 5.5 Programs and Manuals

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

~~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-like 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.~~

### 5.5.9 Secondary Water Chemistry Program

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

- a. ~~Identification of a sampling schedule for the critical variables and control points for these variables;~~
- b. ~~Identification of the procedures used to measure the values of the critical variables;~~
- c. ~~Identification of process sampling points;~~
- d. ~~Procedures for the recording and management of data;~~
- e. ~~Procedures defining corrective actions for all off control point chemistry conditions; and~~
- f. ~~A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.~~

### 5.5.10 Ventilation Filter Testing Program (VFTP)

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

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

| <u>Safety Related Ventilation System</u>          | <u>Flowrate (cfm)</u>       |
|---|-----------------------------|
| Station Emergency Ventilation System (EVS)        | $\geq 7200$ and $\leq 8800$ |
| Control Room Emergency Ventilation System (CREVS) | $\geq 2970$ and $\leq 3630$ |

## 5.5 Programs and Manuals

### 5.5.10 Ventilation Filter Testing Program (continued)

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

| <u>Safety Related Ventilation System</u> | <u>Flowrate (cfm)</u>       |
|--|-----------------------------|
| Station EVS                              | $\geq 7200$ and $\leq 8800$ |
| CREVS                                    | $\geq 2970$ and $\leq 3630$ |

- c. ~~Demonstrate for each of the safety related systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of  $30^{\circ}\text{C}$  ( $86^{\circ}\text{F}$ ) and the relative humidity (RH) specified below:~~

| <u>Safety Related Ventilation System</u> | <u>Penetration (%)</u> | <u>RH (%)</u> |
|--|------------------------|---------------|
| Station EVS                              | $\leq 2.5$             | 95            |
| CREVS                                    | $\leq 2.5$             | 70            |

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

| <u>Safety Related Ventilation System</u> | <u>Delta P<br/>(inches wg)</u> | <u>Flowrate (cfm)</u>       |
|--|--------------------------------|-----------------------------|
| Station EVS                              | $< 6$                          | $\geq 7200$ and $\leq 8800$ |
| CREVS                                    | $< 4.4$                        | $\geq 2970$ and $\leq 3630$ |

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

### 5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

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

## 5.5 Programs and Manuals

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

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the Waste Gas System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and
- b. A surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid storage tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tank's contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

### 5.5.12 Diesel Fuel Oil Testing Program

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

- a. ~~Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:~~
  1. ~~An API gravity or an absolute specific gravity within limits;~~
  2. ~~A flash point and kinematic viscosity within limits for ASTM 2D fuel oil; and~~
  3. ~~A clear and bright appearance with proper color, or a water and sediment content within limits;~~
- b. ~~Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and~~
- c. ~~Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days.~~

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

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

### 5.5.13 Technical Specifications (TS) Bases Control Program

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

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
  1. A change in the TS incorporated in the license; or
  2. A change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.13.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

### 5.5.14 Safety Function Determination Program (SFDP)

~~This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.~~

- a. ~~The SFDP shall contain the following:~~
  1. ~~Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;~~
  2. ~~Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;~~

## 5.5 Programs and Manuals

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### 5.5.14 Safety Function Determination Program (continued)

3. ~~Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and~~
4. ~~Other appropriate limitations and remedial or compensatory actions.~~
- b. ~~A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable; and~~
  1. ~~A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or~~
  2. ~~A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or~~
  3. ~~A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.14.b.1 and 5.5.14.b.2 above is also inoperable.~~
- e. ~~The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.~~

### 5.5.15 Containment Leakage Rate Testing Program

- a. ~~A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(e) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. For Type C tests, this program shall be in accordance with the guidelines contained in Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J," dated July 2012. For Type A and Type B tests, this program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," dated September 1995, as modified by the following exceptions:~~
  1. ~~A reduced duration Type A test may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN TOP-1, Revision 1.~~



## 5.5 Programs and Manuals

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

- ~~2. The fuel transfer tube blind flanges (containment penetrations 23 and 24) will not be eligible for extended test frequencies. Their Type B test frequency will remain at 30 months. However, as found testing will not be required.~~
- ~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is 38 psig.~~
- ~~c. The maximum allowable containment leakage rate,  $L_a$ , at  $P_a$ , shall be 0.50% of containment air weight per day.~~
- ~~d. Leakage rate acceptance criteria are:~~
  - ~~1. Containment leakage rate acceptance criterion is  $< 1.0 L_a$ . During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are  $< 0.60 L_a$  for the Type B and G tests and  $\leq 0.75 L_a$  for Type A tests.~~
  - ~~2. Air lock testing acceptance criteria are:~~
    - ~~a) Overall air lock leakage rate is  $\leq 0.015 L_a$  when tested at  $\geq P_a$ .~~
    - ~~b) For each door, leakage rate is  $\leq 0.01 L_a$  when the volume between the door seals is pressurized to  $\geq 10$  psig.~~
- ~~e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

### 5.5.16 Battery Monitoring and Maintenance Program

~~This Program provides for battery restoration and maintenance, including the following:~~

- ~~a. Actions to restore battery cells with float voltage  $< 2.13$  V;~~
- ~~b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; and~~
- ~~c. Actions to verify that the remaining cells are  $> 2.07$  V when a pilot cell or cells have been found to be  $< 2.13$  V.~~

### 5.5.17 Control Room Envelope Habitability Program

~~A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE~~

## 5.5 Programs and Manuals

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### 5.5.17 Control Room Envelope Habitability Program (continued)

~~occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:~~

- ~~a. The definition of the CRE and the CRE boundary;~~
  - ~~b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance;~~
  - ~~c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Section C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0;~~
  - ~~d. Measurements, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary;~~
  - ~~e. The quantitative limits on unfiltered air leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in Specification 5.5.17.c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis; and~~
  - ~~f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by Specifications 5.5.17.c and 5.5.17.d, respectively.~~
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## 5.0 ADMINISTRATIVE CONTROLS

### 5.6 Reporting Requirements

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

#### 5.6.1 Annual Radiological Environmental Operating Report

**facility** The Annual Radiological Environmental Operating Report covering the operation of the ~~unit~~ during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

#### 5.6.2 Radioactive Effluent Release Report

**facility** The Radioactive Effluent Release Report covering the operation of the ~~unit~~ in the previous year shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the ~~unit~~. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

#### 5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. ~~Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:~~

1. ~~SL 2.1.1.1, "Reactor Core Safety Limits";~~
2. ~~LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";~~
3. ~~LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";~~
4. ~~LCO 3.1.7, "Position Indicator Channels," (SR 3.1.7.1 limits);~~

## 5.6 Reporting Requirements

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### 5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. ~~LCO 3.1.8, "PHYSICS TEST Exceptions MODE 1";~~
  6. ~~LCO 3.1.9, "PHYSICS TEST Exceptions MODE 2";~~
  7. ~~LCO 3.2.1, "Regulating Rod Insertion Limits";~~
  8. ~~LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";~~
  9. ~~LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits";~~
  10. ~~LCO 3.2.4, "QUADRANT POWER TILT (QPT)";~~
  11. ~~LCO 3.2.5, "Power Peaking Factors";~~
  12. ~~LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation,"  
Function 8 (Flux  $\Delta$ Flux Flow) Allowable Value; and~~
  13. ~~LCO 3.9.1, "Boron Concentration."~~
- b. ~~The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in BAW 10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," or any other new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW 10179P-A. The applicable approved revision number for BAW 10179P-A at the time of the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT (COLR). The COLR shall also list any new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW 10179P-A.~~
- e. ~~As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of RTP is specified in a previously approved method, an actual value of 100.37% of RTP may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:~~
1. ~~Caldon Inc. Engineering Report 80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM<sup>TM</sup> System," Revision 0, dated March, 1997.~~

## 5.6 Reporting Requirements

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### 5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. ~~Caldon Inc. Engineering Report 157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM<sup>√</sup>™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.~~
- d. ~~The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~
- e. ~~The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.~~

### 5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. ~~RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~
  1. ~~LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."~~
- b. ~~The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:~~
  1. ~~BAW 10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," June 1986;~~
  2. ~~ASME Code Section XI, Appendix G, 1995 Edition with Addenda through 1996, as modified by the alternative procedures provided in ASME Code Case N-640 and ASME Code Case N-588; and~~
  3. ~~BAW 2308, Revision 1 A and Revision 2 A, "Initial RT<sub>NDT</sub> of Linde 80 Weld Materials," August 2005 and March 2008, respectively.~~
- e. ~~The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.~~

## 5.6 Reporting Requirements

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### 5.6.5 Post Accident Monitoring Report

~~When a report is required by Condition B or F of LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.~~



## 5.6 Reporting Requirements

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### 5.6.6 Steam Generator Tube Inspection Report

~~A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, "Steam Generator (SG) Program." The report shall include:~~

- ~~a. The scope of inspections performed on each SG;~~
- ~~b. 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;~~
- ~~g. The results of condition monitoring, including the results of tube pulls and in-situ testing;~~

### 5.6.7 Remote Shutdown System Report

~~When a report is required by Condition C of LCO 3.3.18, "Remote Shutdown System," a report shall be submitted within the following 30 days. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the control circuit or transfer switch of the Function to OPERABLE status.~~

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## Attachment 2

### Technical Specification Bases Page Markups (for information only)

(19 pages follow)

Technical Specification Bases that are deleted in their entirety are identified as such in the Technical Specification Bases Table of Contents, however, the associated deletions are not included in this attachment. The remaining Technical Specification Bases are intentionally not re-numbered.

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## B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

## BASES

|           |   |
|-----------|---|
| LCOs      | LCO 3.0.1 <del>through LCO 3.0.8</del> <b>and 3.0.2</b> 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 <b>facility</b> unit is in the <del>MODES or other</del> specified conditions of the Applicability statement of each Specification).   |
| LCO 3.0.2 | <p>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:</p> <ul style="list-style-type: none"> <li>a. <del>Completion</del> of the Required Actions within the specified Completion Times constitutes compliance with a Specification; <b>and</b> <del>.</del> <b>completion</b></li> <li>b. <del>Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.</del></li> </ul> <p><del>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.</del></p> <p><del>Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.</del></p> |



## BASES

~~LCO 3.0.2 (continued)~~

~~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. Reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. 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.~~

~~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.~~

[NOTE: The Bases for LCOs 3.0.3 through 3.0.8 are deleted (pages B 3.0.3 through B 3.0-13)]

## 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 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.   |
| SR 3.0.1 | <p>SR 3.0.1 establishes the requirement that SRs must be met during the <del>MODES or other</del> 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 <del>the OPERABILITY of systems and components, and</del> 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 may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. <del>Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.</del></p> <p><del>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:</del></p> <ul style="list-style-type: none"> <li><del>a. The systems or components are known to be inoperable, although still meeting the SRs; or</del></li> <li><del>b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.</del></li> </ul> <p><del>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 LCO are only applicable when the Test Exception LCO is used as an allowable exception to the requirements of a Specification.</del></p> <p><del>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.</del></p> |

Variables are assumed to be within limits

variables are within limits

the

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BASES

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## 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 status.~~

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with 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.~~

~~Some examples of this process are:~~

- ~~a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the AFW pump testing.~~
- ~~b. Main steam safety valve (MSSV) lift setpoint verification performed in-situ requires hot conditions. Provided other appropriate ANSI/ASME OM Code test requirements are satisfactorily completed, startup can proceed and MODE 3 entered with the MSSVs considered OPERABLE. This allows operation to reach the necessary conditions to perform the in-situ lift setpoint verification.~~

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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.~~

~~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).~~

facility →

BASES

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## SR 3.0.2 (continued)

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.2 are applicable, 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.

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. ~~The requirements of regulations take precedence over the TS. Examples of where SR 3.0.2 does not apply are the Containment Leakage Rate Testing Program required by 10 CFR 50, Appendix J, and the inservice testing of pumps and valves in accordance with applicable American Society of Mechanical Engineers Operation and Maintenance Code, as required by 10 CFR 50.55a. These programs establish testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations directly or by reference.~~

~~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.~~

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.

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SR 3.0.3

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.

BASES

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## SR 3.0.3 (continued)

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

This delay period provides an adequate time to perform Surveillances that have been missed. 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.

facility



The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

facility



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~~

## BASES

## SR 3.0.3 (continued)

variables are  
within limits

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. 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 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 risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, 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



## BASES

## SR 3.0.3 (continued)

~~use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.~~

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.

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. ~~another~~

limits ensure  
facility safety

variable limits

~~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 a Surveillance not being met in accordance with LCO 3.0.4.~~

~~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~~

BASES

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## SR 3.0.4 (continued)

~~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 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 may (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 entry into 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 into 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."~~

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## B 3.7 PLANT SYSTEMS

### B 3.7.14 Spent Fuel Pool Water Level

#### BASES

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|                            |   |
|----------------------------|---|
| BACKGROUND                 | <p>The minimum water level in the spent fuel pool meets the assumption 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.</p> <p>A general description of the spent fuel pool design is given in the UFSAR, Section 9.1.2, Reference 1. The Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.4.7 (Ref. 3).</p>   |
| APPLICABLE SAFETY ANALYSES | <p>The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is below 10 CFR 100 (Ref. 5) guidelines.</p> <p>According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With 23 ft, the assumptions of Reference 4 can be used directly. In practice, the LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel rack, however, there may be &lt; 23 ft above the top of the fuel bundle and the surface, by the width of the bundle. The fuel handling accident assumes the entire outer row of fuel rods in the assembly, 56 fuel rods out of 208 total fuel rods, suffer mechanical damage to the cladding.</p> <p>The spent fuel pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).</p> |
| LCO                        | <p>The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for irradiated fuel movement within the spent fuel pool.</p>  |
| APPLICABILITY              | <p>This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool since the potential for a release of fission products exists.</p>  |

BASES

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## ACTIONS

A.1

When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the spent fuel pool at less than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of loads over the spent fuel. This does not preclude movement of a fuel assembly to a safe position.

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. 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, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

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SURVEILLANCE  
REQUIREMENTSSR 3.7.14.1

This SR verifies that sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

~~During refueling operations, the level in the spent fuel pool is at equilibrium with that in the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.~~

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## REFERENCES

1. UFSAR, Section 9.1.2.
  2. UFSAR, Section 9.1.3.
  3. UFSAR, Section 15.4.7.
  4. Regulatory Guide 1.25.
  5. 10 CFR 100.11.
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## B 3.7 PLANT SYSTEMS

### B 3.7.15 Spent Fuel Pool Boron Concentration

#### BASES

|                                  |   |
|----------------------------------|---|
| BACKGROUND                       | <p>As described in LCO 3.7.16, "Spent Fuel Pool Storage," fuel assemblies are stored in the spent fuel pool racks in a Mixed Zone Three Region, Checkerboard, or Homogenous Loading pattern in accordance with criteria based on initial enrichment and assembly burnup. The high density spent fuel pool storage racks in the Spent Fuel Pool (SFP) are designed to assure that the effective neutron multiplication factor, <math>k_{\text{eff}}</math>, is <math>\leq 0.95</math> with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water.</p>  |
| APPLICABLE<br>SAFETY<br>ANALYSES | <p>Reactivity effects of abnormal and accident conditions have been evaluated to assure that under credible abnormal and accident conditions, the reactivity will not exceed 0.95, with credit for soluble boron in the pool water. Assuring the presence of soluble poison during fuel handling operations precludes the possibility of the simultaneous occurrence of two independent accident conditions.</p> <p>Three potential accident scenarios, misloaded fresh fuel assembly, mislocated fresh fuel assembly, and a dropped fuel assembly, were analyzed to determine the effect the accidents would have on the effective neutron multiplication factor, <math>k_{\text{eff}}</math>. The results of the analysis determined that a minimum boron concentration of 630 ppm in the SFP water is required to maintain <math>k_{\text{eff}} \leq 0.95</math> for the worst-case accident scenario (i.e., a 5.05 weight percent enriched fresh fuel assembly misloaded in a Checkerboard pattern) (Ref. 1). The minimum boron concentration value of 630 ppm bounds all analyzed potential accident scenarios discussed below.</p> <p>A misloaded fresh fuel assembly accident scenario analyzed misloading the assembly in the following five different locations: 1) misloading in the Mixed Zone Three Region (MZTR) inner rack 10x9; 2) misloading in the MZTR inner rack 10x9 (different location of a fresh assembly); 3) misloading in the MZTR side rack 10x8; 4) misloading in Homogeneous (45 BU) inner rack 10x9, and; 5) misloading in Checkerboard inner rack 10x9. The worst case scenario, misloading in Checkerboard inner rack 10x9, requires a minimum boron concentration of 627 ppm to assure that <math>k_{\text{eff}}</math> does not exceed 0.95.</p> <p>The second potential accident scenario considers the mislocation of a fresh fuel assembly outside of a storage rack adjacent to other fuel assemblies. The worst case would be an assembly mislocated in a corner on the west side of the pool (next to MZTR outer rack 10x8 – 7x1). This scenario requires a minimum boron concentration of 448 ppm to assure that <math>k_{\text{eff}}</math> does not exceed 0.95.</p> |

## BASES

## APPLICABLE SAFETY ANALYSES (continued)

The dropped fuel assembly accident considers three different scenarios: a dropped fuel assembly coming to rest horizontally on top of the rack; a dropped fuel assembly came to rest vertically into a location occupied by another assembly, and; dropping the fuel assembly into an unoccupied cell. In all cases, a minimum boron concentration of 53 ppm is adequate to assure that  $k_{\text{eff}}$  does not exceed 0.95.

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

## LCO

The specified concentration  $\geq 630$  ppm of dissolved boron in the spent fuel pool preserves the assumption used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

## APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. 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 movement 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 pool 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 the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. The concentration of boron is restored simultaneously with suspending movement of the fuel assemblies. Alternatively, beginning a verification of the spent fuel pool 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.

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. 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 MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.~~



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

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REFERENCES

1. UFSAR, Section 9.1.2.1.
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## B 3.7 PLANT SYSTEMS

## B 3.7.16 Spent Fuel Pool Storage

## BASES

|                            |  |
|----------------------------|--|
| BACKGROUND                 | <p>The spent fuel storage facility is designed to store <del>either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies</del> in a vertical configuration underwater. The high density spent fuel pool storage racks are designed to maintain a <math>k_{\text{eff}}</math> equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for manufacturing tolerances and calculation uncertainty. The spent fuel pool facility is designed to assure the safe storage of irradiated fuel assemblies under normal and accident conditions. Each storage rack consists of a rectangular array of stainless steel cells with walls of 0.075 inches nominal thickness, spaced a nominal 9.22 inches on center in both directions. The neutron absorber material is utilized between each cell for criticality considerations. The 21 spent fuel pool racks store a maximum of 1624 fuel assemblies. The rack cells are arranged in parallel rows with a center-to-center spacing of 9.22 inches.</p>   |
| APPLICABLE SAFETY ANALYSES | <p>The spent fuel storage facility is designed for noncriticality by use of adequate spacing. A neutron absorber is attached to all four sides of each cell. In addition, there is a gap between individual racks and between the peripheral racks and the pool walls. These gaps form flux traps that reduces neutron movement between fuel assemblies in adjacent racks. Loading patterns maintain <math>k_{\text{eff}} &lt; 0.95</math> for fuel assemblies with initial nominal enrichments <math>\leq 5.05</math> weight percent Uranium-235, assuming the spent fuel pool water is unborated.</p> <p>The spent fuel pool storage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).</p>  |
| LCO                        | <p>The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 3.7.16-1, ensure that the <math>k_{\text{eff}}</math> of the spent fuel pool will always remain <math>&lt; 0.95</math> assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool, according to Figure 3.7.16-1. The restrictions on the placement of fuel assemblies within the spent fuel pool as dictated by Figure 3.7.16-1 ensure that the <math>k_{\text{eff}}</math> of the spent fuel pool will always be <math>&lt; 0.95</math> assuming the spent fuel pool is flooded with non-borated water. The restrictions delineated in Figure 3.7.16-1 and the Required Actions are consistent with the criticality safety analysis performed for the spent fuel pool (Ref. 1).</p> <p>The criticality analyses qualify the high density rack modules for storage of the fuel assemblies in one of three different loading patterns subject to certain restrictions: Mixed Zone Three Region (MZTR), Checkerboard (CB), and Homogeneous Loading (HL). Figure 3.7.16-1 provides the Category-specific burnup/enrichment limitations. Different loading</p> |

## BASES

## LCO (continued)

patterns may be used in different rack modules, provided each rack module contains only one loading pattern. Two different loading patterns may be used in a single rack module, subject to certain additional restrictions. The loading pattern restrictions are maintained in fuel handling administrative procedures.

MZTR is a loading pattern where ~~fresh or~~ low burnup assemblies (identified as Region 1 assemblies) are separated from each other and from intermediate burnup fuel assemblies (identified as Region 3 assemblies) by barrier fuel assemblies with high burnup (identified as Region 2 assemblies). CB is a loading pattern of empty cells, or cells with non-fuel bearing components, and cells with ~~fresh or~~ low burnup assemblies (Region 1). HL is a loading pattern of intermediate burnup fuel assemblies (Region 3). Region 2 assemblies correspond to Category A in Figure 3.7.16-1, Region 3 assemblies correspond to Category B in Figure 3.7.16-1, and Region 1 assemblies correspond to Category C in Figure 3.7.16-1. ← New fuel is no longer stored onsite.

|                           |  |
|---------------------------|--|
| APPLICABILITY             | This LCO applies whenever any fuel assembly is stored in the spent fuel pool.  |
| ACTIONS                   | <p><u>A.1</u></p> <p>When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.16-1.</p> <p><del>Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.</del></p> |
| SURVEILLANCE REQUIREMENTS | <p><u>SR 3.7.16.1</u></p> <p>This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1.</p>   |
| REFERENCES                | 1. UFSAR, Section 9.1.2.1.   |