Southern Nuclear

Cheryl A. Gayheart
Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5316 tel
205 992 7601 fax
cagayhea@southernco.com

September 30, 2019

10 CFR 50.90

Docket Nos.: 50-348
50-364

NL-19-0796

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

Joseph M. Farley Nuclear Plant Units 1 and 2
Submittal of License Amendment Request (LAR) to
Update the Spent Fuel Pool Criticality Safety Analysis and Revise
Technical Specification (TS) 3.7.15 "Spent Fuel Assembly Storage" and TS 4.3 "Fuel Storage"

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 Renewed Facility Operating License (NPF-2), and Unit 2 Renewed Facility Operating License (NPF-8), by incorporating the attached proposed change into the Unit 1 and Unit 2 Technical Specifications (TSs). Specifically, the proposed change is a request to revise TS 3.7.15 and TS 4.3 to:

- allow for an updated spent fuel pool (SFP) criticality safety analysis; and
- account for the impact on the spent fuel from a proposed measurement uncertainty recapture (MUR) power uprate.

The Enclosure provides a description and assessment of the proposed changes. Attachment 1 provides the existing TS pages marked to show the proposed changes. Attachment 2 provides retyped TS pages. Attachment 3 provides existing TS Bases pages marked to show the proposed changes for information only. Attachment 4 provides a proprietary version of the technical evaluation [WCAP]. Attachment 5 provides a non-proprietary version of the technical evaluation [WCAP]. Attachment 6 provides the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-19-4943, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

Approval of the proposed amendments is requested by October 1, 2020 to support the Fall 2020 refueling outage for FNP Unit 2. The proposed changes will be implemented within 90 days of issuance of the amendment.

The Fall 2020 refueling outage is the implementation date for the FNP MUR power uprate modifications. SNC plans to submit the MUR LAR by the end of October 2019.

A001
NRR

This letter contains no NRC commitments.

In accordance with 10 CFR 50.91, SNC is notifying the state of Alabama of this license amendment request by transmitting a copy of this letter to the designated state official.

If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 30, 2019.

C. A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Company

efb/scm

Enclosure: Description and Assessment of the Proposed Changes
Attachment 1: Technical Specification Page Markups
Attachment 2: Retyped Technical Specification Pages
Attachment 3: Technical Specification Bases Page Markups – For Information only
Attachment 4: WCAP-18414-P "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary Version)
Attachment 5: WCAP-18414-NP "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Non-proprietary Version)
Attachment 6: Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-19-4943, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

cc: NRC Regional Administrator
NRC NRR Project Manager – Farley 1&2
NRC Senior Resident Inspector – Farley 1 & 2
Alabama - State Health Officer for the Department of Public Health
SNC Document Control R-Type: CFA04.054

ENCLOSURE

Description and Assessment of the Proposed Changes

Subject: Joseph M. Farley Nuclear Plant Units 1 and 2 Submittal of License Amendment Request to Revise Technical Specification (TS) 3.7.15 "Spent Fuel Assembly Storage" and TS 4.3 "Fuel Storage"

ATTACHMENTS
1. Technical Specification Page Markups
2. Retyped Technical Specification Pages
3. Technical Specification Bases Page Markups – For Information Only
4. WCAP-18414-P "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary Version)
5. WCAP-18414-NP "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Non-proprietary Version)
6. Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-19-4943, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

1. SUMMARY DESCRIPTION

Southern Nuclear Operating Company (SNC) requests an amendment to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 Renewed Facility Operating License (NPF-2) and Unit 2 Renewed Facility Operating License (NPF-8) by incorporating the attached spent fuel storage changes into the FNP Unit 1 and Unit 2 Technical Specifications. The license amendment request proposes changes to spent fuel storage Technical Specification (TS) 3.7.15 "Spent Fuel Assembly Storage" and TS 4.3 "Fuel Storage" for FNP Unit 1 and Unit 2. The purpose of the proposed changes is to allow for an updated spent fuel pool criticality safety analysis and to account for the impact on the spent fuel from a proposed measurement uncertainty recapture (MUR) power uprate.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

The spent fuel pool (SFP) is made up of one fuel storage rack design (region) that maintains 10.75-inch center to center spacing between spent fuel assemblies. The Farley Units 1 & 2 SFPs each consist of two 6 x 7, nineteen 7 x 7, and seven 7 x 8 storage racks. The spent fuel racks are freestanding and are free to move on the pool liner floor during a seismic event.

The SFP storage capacity is 1,407 fuel assemblies. The actual storage capacity is limited by Technical Specification 3.7.15, Spent Fuel Assembly Storage, and is dependent upon the fuel characteristics of the FNP fuel inventory.

The storage racks are of flux trap style with an uncredited Boraflex neutron absorber panel on the sides of each storage cell. This results in a flux trap between any two assembly storage locations. No credit is taken for the presence of residual Boraflex in the current licensing basis per Section 4.3.2.7.2.1 and 4.3.2 7.2.2 of the FNP Units 1 and 2 Updated Final Safety Analysis Report (UFSAR). The burnable absorber cavity is assumed to be filled with water of the same composition as the water elsewhere in the storage racks.

Per Section 9.1.2.1 of the FNP Units 1&2 UFSAR, the spent fuel racks are designed to withstand shipping, handling, and normal operating loads (impact and dead loads of fuel assemblies) as well as safe shutdown earthquake (SSE) and one-half SSE seismic loads meeting American Nuclear Society (ANS) Safety Class 3 and American Institute of Steel Construction (AISC) requirements. The spent fuel racks are also designed to meet the Category 1 seismic requirements of Regulatory Guide 1.13.

The updated criticality safety analysis, "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Attachment 4 - WCAP-18414-P, Proprietary Version and Attachment 5 - WCAP-18414-NP, Non-Proprietary Version) evaluates the SFP storage racks for the placement of fuel within the storage arrays defined in the technical specifications. Credit is taken for the negative reactivity associated with burnup and post-irradiation cooling time (decay time) for assemblies which have been operated in the reactor. Fuel assemblies which have not operated in the reactor may take credit for the presence of zirconium diboride in the integral fuel burnable absorber (IFBA). While the FNP Unit 1 and Unit 2 SFP storage racks may contain Boraflex absorber inserts, no credit is taken for the presence of Boraflex absorber as described in the UFSAR. However, credit is taken for the presence of soluble boron in the SFP.

2.2 Current Technical Specifications Requirements

SNC proposes changes to the spent fuel storage and affects Technical Specification (TS) 3.7.15 "Spent Fuel Assembly Storage" and TS 4.3 "Fuel Storage" for FNP Unit 1 and Unit 2.

2.3 Reason for the Proposed Change

The purpose of the proposed changes to TS 3.7.15 and TS 4.3 is to update the SFP criticality safety analysis. The updated analysis, per the NRC's request, follows the guidance in NEI 12-16 (Reference 6.1). The proposed changes also account for the impact on the spent fuel from a proposed measurement uncertainty recapture (MUR) power uprate.

2.4 Description of the Proposed Change

TS 3.7.15, "Spent Fuel Assembly Storage"

The proposed change revises TS 3.7.15 based on the updated criticality safety analysis, "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Attachment 4 - WCAP-18414-P, Proprietary Version and Attachment 5 - WCAP-18414-NP, Non-Proprietary Version) and the proposed changes to TS 4.3.

The proposed changes to TS 3.7.15 include:

- removal of the reference to "within the Acceptable Burnup Domain of Figure 3.7.15-1",
- removal of the reference to "Figure 3.7.15-1 or" in Surveillance Requirement (SR) 3.7.15.1, and
- removal of TS Figure 3.7.15-1 which contains the fuel assembly burnup limit requirements for all cell storage. This figure will be replaced by Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5 in Section 4.3.

Note: Attachment 4 and Attachment 5 are applicable to all references to the updated criticality safety analysis from hereon.

TS 4.3, "Fuel Storage"

The proposed change to TS 4.3 revises the section based on the updated criticality safety analysis. The proposed changes include updated fuel storage configurations (arrays) and restrictions for storage of fuel in the storage racks based on fuel assembly burnup limit requirements.

The proposed changes to TS 4.3 include:

- Replacement of Specification 4.3.1.1.e with "New or partially spent fuel assemblies that must be stored according to their combination of discharge burnup and nominal enrichment, decay time since operation, required IFBA (if applicable), and must comply with Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5 (as applicable). Each assembly should be stored in an appropriate storage configuration according to its fuel category as specifically described in Table 4.3-1 and geometry based on Figure 4.3-1;"
- Replacement of Specification 4.3.1.1.f with "Fuel assemblies that are stored in accordance with every applicable storage array as shown in Figure 4.3-1 of which they

are a part (i.e., one fuel assembly can be part of up to four different storage arrays, each storage array shall be in accordance with Figure 4.3-1); and"

- Replacement of "Figure 4.3-6" with "Figure 4.3-2" in Specification 4.3.1.1.g.
- Replacement of Figures 4.3-1 through 4.3-5 with proposed Figure 4.3-1 and Tables 4.3-1 through 4.3-5.
    - The proposed Figure 4.3-1 contains graphical and verbal descriptions of the four allowed storage arrays.
    - The proposed Table 4.3-1 provides interpretation of the fuel categories included in Figure 4.3-1 and it includes references to the appropriate table that provides the fitting coefficients that are to be used to calculate the minimum required fuel assembly burnup or the minimum IFBA requirement.
    - The proposed Table 4.3-2 provides the maximum enrichment allowed for each fuel category with 0.0 MWd/MTU burnup.
    - The proposed Table 4.3-3 provides the fitting coefficients to calculate the minimum required fuel assembly burnup for fuel categories 3 and 4 for Standard Fuel Assembly (STD)/Robust Fuel Assembly (RFA) fuel.
    - The proposed Table 4 3-4 provides the fitting coefficients to calculate the minimum required fuel assembly burnup for fuel categories 3 and 4 for Optimized Fuel Assembly (OFA) fuel.
    - The proposed Table 4.3-5 provides the fitting coefficients to calculate the minimum IFBA requirements for fuel category 2.
- Renumbering Figure 4.3-6 as Figure 4.3-2. The specifications that reference Figure 4.3-6 are being changed to reference Figure 4.3-2.

Redline/strikeout copies of TS 3.7.15 and TS 4.3 are included in Attachment 1. Retyped copies TS 3.7.15 and TS 4.3 are provided in Attachment 2. Redline/strikeout copies of TS 3.7.14 Bases and TS 3.7.15 Bases are provided (for information only) in Attachment 3.

3. TECHNICAL EVALUATION

This License Amendment Request, at the NRC's request, was modeled based on the guidance of NEI 12-16 (Reference 6.1). SNC used the NEI guidance to ensure that the proper considerations were made in the analysis and controls. The updated criticality safety analysis also reflects the guidance of DSS-ISG-2010-001, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pool" (Reference 6.2).

This section includes a brief statement related to each applicable topic discussed in NEI 12-16 (Reference 6.1) and summarizes the analysis or proposed controls applicable to each area. Also included is a discussion on the proposed Technical Specification changes. Supporting details are found in Attachments 4/5, as referenced below.

<u>Acceptance Criteria</u>

NEI 12-16, Section 2 (Reference 6.1) describes the NRC acceptance criteria for spent fuel pool storage of new and used fuel for pools where credit for soluble boron is taken as follows:

1. The criticality safety analyses must meet two independent limits:

a. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level.

b. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with borated water, the k_{eff} must not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

Section 2.1 of the updated criticality safety analysis in Attachment 4 provides the same acceptance criteria for the updated FNP SFP criticality analysis and carries forward the criteria approved in the NRC Safety Evaluation issued with the FNP SFP criticality analysis WCAP-14416-P approved in Amendment No. 133 to Facility Operating License No. NPF-2 and Amendment No. 125 to Facility Operating License No. NPF-8 for Joseph M. Farley Nuclear Plant, Units 1 and 2 (Accession No. ML013130226).

Computer Codes

NEI 12-16, Section 3 (Reference 6.1) describes the different types of computer codes that may be used in a criticality analysis. This section also discusses the validation of the computer codes used in the criticality analysis. The licensee needs to state which codes were utilized along with the type/version of cross-section libraries.

The updated criticality safety analysis utilized in this submittal employs the following computer codes and cross-section libraries:

(1) The two-dimensional (2-D) transport lattice code PARAGON Version 1.2.0 and its cross-section library based on Evaluated Nuclear Data File Version VI.3 (ENDF/B-VI.3).

(2) Scale Version 6.2.3 with the ENDF/B-VII 238-group cross-section library.

The Computer Codes used in this application are discussed in Section 2.3 of the updated criticality safety analysis. Paragon is generically approved for depletion calculations. The applicable specific validation of Scale Version 6.2.3 is provided in Appendix A of the updated criticality safety analysis.

Reactivity Effects of Depletion

NEI 12-16, Section 4 (Reference 6.1) describes appropriate considerations for calculating reactivity effects of fuel depletion. Significant parameters that could impact reactivity of used fuel in depletion analyses are:

- Power, moderator temperature and fuel temperature during depletion
- Soluble boron during depletion
- Presence of burnable absorbers
- Rodded operation

Cooling time and other depletion parameters are also discussed in NEI 12-16, Section 4 (Reference 6.1).

The depletion analysis is described in Section 4 of the updated criticality safety analysis and describes the methods used to determine conservative and bounding inputs for the generation of isotopic number densities. Controls which ensure future fuel designs satisfy the assumptions of the analysis are discussed in the Licensee Controls section, below. Depletion uncertainty is discussed in Section 5.2.3.1.5 of the updated criticality safety analysis.

Fuel Assembly and Storage Rack Modeling

NEI 12-16, Section 5 (Reference 6.1) describes generally acceptable methods of modeling fuel assemblies and fuel storage racks, including considerations for rack neutron absorbers.

Two fuel assembly designs are used at Farley and are considered in the updated criticality safety analysis, the Westinghouse standard fuel assembly (STD) and the Westinghouse optimized fuel assembly (OFA). The Westinghouse robust fuel assembly (RFA) is also considered in the updated criticality safety analysis to support potential future use. The burnup requirements developed for the STD fuel design are applicable to the RFA fuel design because the neutronically important characteristics are the same. Details of the fuel assembly designs are provided in Section 3.1 and 4.3.1 of the updated criticality safety analysis. The updated criticality safety analysis allows fuel assemblies which have not operated in the reactor to take credit for the presence of zirconium diboride in the integral fuel burnable absorber (IFBA).

The spent fuel pool is made up of one fuel storage rack design (region). The Farley Units 1 & 2 SFPs each consist of two 6 x 7, nineteen 7 x 7, and seven 7 x 8 storage racks. The storage racks are of flux trap style with an uncredited Boraflex neutron absorber panel on every side of each storage cell. This results in a flux trap between any two assembly storage locations. No credit is taken for the presence of residual Boraflex. The burnable absorber cavity is assumed to be filled with water of the same composition as the water elsewhere in the storage racks. Credit is taken for the presence of soluble boron in the SFP. Details of the storage rack parameters are provided in Section 3.2 of the updated criticality safety analysis.

The fuel and storage rack manufacturing tolerances, eccentric fuel assembly positioning bias, and SFP temperature bias are included in the updated criticality safety analysis through either analysis or use of bounding values. Details of these items are provided in Section 5.2.3.1 of the updated criticality safety analysis

The axial burnup distribution and reactor record burnup uncertainties are considered in the updated criticality safety analysis. Details of the axial burnup distribution are provided in Section 4.2.3. The details for the burnup uncertainties are provided in Section 5.2.3.1.4.

The new fuel storage racks were not considered in the updated SFP criticality safety analysis update. This license amendment involves no changes to the design or operation of the new fuel storage racks.

Configuration Modeling and Soluble Boron Credit

NEI 12-16, Section 6 (Reference 6.1) describes considerations for configuration modeling, including a description of normal conditions, interface considerations, and abnormal / accident conditions which should be considered. NEI 12-16, Section 7 (Reference 6.1) describes considerations for soluble boron credit under normal and accident conditions, and considerations for a boron dilution accident.

The updated criticality safety analysis demonstrates acceptable results for k_{eff} for both normal conditions (Section 5.4) and accident conditions (Section 5.5.2). Normal conditions include normal storage, fuel movement, and other procedurally controlled activities in the SFPs. Fuel assembly storage arrays which define allowable storage are defined in Section 5.2.1 of the updated criticality safety analysis, which have been incorporated into the proposed Technical Specification changes. Controls which ensure that the proposed Technical Specification limitations for storage are maintained are discussed in the Licensee Controls section below.

Interface considerations are described in Section 5.3 of the updated criticality safety analysis. The interfaces are the locations where there is a change in either the storage racks or the storage requirements of the fuel in question. Only the intra-region interfaces are evaluated because all racks are of the same design and no pool region interfaces are present. Each storage cell can be part of four different storage arrays. Compliance with the storage arrays in the technical specifications will ensure acceptable boundary cells at the interface.

Per Section 4.3.2.7.2.1 of the FNP Units 1&2 UFSAR, most postulated accidents in the spent-fuel rack will not result in an increase in reactivity. These include dropping an assembly on top of the rack (rack structure maintains 10 in. of separation between dropped and stored assemblies, precluding interaction), or dropping an assembly into a position other than a storage cell (prevented by design of rack). However, accidents can be postulated for each storage configuration which would increase reactivity beyond the analyzed condition. The first postulated accident would be a loss of the fuel pool cooling system. The second accident would be dropping an assembly into an already loaded cell and the third would be a misload of an assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied.

Single misload events were previously analyzed; however, the updated criticality safety analysis also includes analysis of a multiple misload accident scenario in accordance with NEI 12-16, Section 6.3.5. The inclusion of this analysis does not imply the creation of the possibility of a new accident, but expands the boundaries of the analyzed accident conditions to ensure that all potential accidents are properly considered.

For the limiting normal condition, 320 ppm of soluble boron is credited to ensure the maximum k_{eff} satisfies the acceptance criteria of $k_{eff} \leq 0.95$.

The limiting analyzed accident condition was an event which involves misloading multiple fuel assemblies in series due to a common cause.

In the multiple misload accident cases, 1710 ppm of boron is required to maintain $k_{eff} \leq 0.95$. This amount of boron is bounded by the current limit in Technical Specification 3.7.14 "Fuel Storage Pool Boron Concentration" which requires greater than 2000 ppm of boron concentration in the fuel storage pool.

A spent-fuel pool dilution event has been previously evaluated by SNC and determined to not be a credible event for Farley Units 1 & 2. Section 4.3.2.7.2.1 of the FNP Units 1&2 UFSAR provides the following summary:

> "A spent-fuel pool boron dilution evaluation was performed to determine the volume necessary to dilute the spent-fuel pool from the Technical Specification limit of 2000 ppm to 400 ppm (the boron concentration required to maintain $K_{eff} \leq 0.95$). The boron dilution evaluation determined that approximately 480,000 gal of water would be required to dilute

> the spent-fuel pool from 2000 ppm to 400 ppm. A dilution event that would result in this large volume of water would require the transfer of a large quantity of water from the dilution source and a significant increase in the spent-fuel pool level, which would ultimately overflow the pool.
>
> This large volume of water would be readily detected and terminated by plant personnel. A spent-fuel pool dilution event of this magnitude is not a credible event."

SNC's previous evaluation was considered and accepted in the NRC Safety Evaluation for Amendments 133/125, and therefore, the spent-fuel pool dilution event was not re-analyzed as part of the updated criticality safety analysis.

The abnormal/accident section of NEI 12-16, Section 6 (Reference 6.1) also includes seismic events. Section 5.5.2.4 of the updated criticality safety analysis addresses the seismic event. In the event of an earthquake or similar seismic event, the SFP storage racks can shift position. This can cause the rack modules to slide together eliminating the space between modules and between modules and the spent fuel pool wall. The fuel assembly position analysis in the updated criticality safety analysis covers possible water gap reduction between assemblies due to a seismic event. Similar to the multiple misload accident scenario, the inclusion of this analysis did not imply the creation of the possibility of a new accident but expands the boundaries of the analyzed accident conditions to ensure that all potential accidents are properly considered. The effects of this event are bounded by the worst-case fuel assembly misloading event.

<u>Calculation of Maximum k_{eff}</u>

NEI 12-16, Section 8 (Reference 6.1) describes that the maximum k_{eff} is determined by adding to the nominal calculated k_{eff} any biases that may exist in the methodology and the applicable uncertainties using the formula described below, for comparison to the acceptance limits.

$$k_{max} = k_{eff} + \sum_{i=0}^{m} Bias_i + \sqrt{\sum_{j=0}^{n} Uncertainty_j^2}$$

The updated criticality safety analysis demonstrates that the k_{eff}, including all applicable biases and uncertainties which account for the statistical 95/95 confidence levels, satisfy the acceptance criteria. The sum of biases are additive while the sum of uncertainties are statistically added as the root sum square of the individual reactivity uncertainties as described in Sections 5.2.2 and 5.2.3 of the updated criticality safety analysis.

<u>Licensee Controls</u>

NEI 12-16, Section 9 (Reference 6.1) describes controls intended to ensure that the conditions evaluated in the nuclear criticality safety analysis are and remain bounding to the current plant operating parameters. It discusses procedural controls for fuel storage and for planning and performance of fuel movements, new (future) fuel types, and pre- and post-irradiation fuel characterization.

In conjunction with the implementation of the approved LAR, the controls are changed to be consistent with the updated criticality safety analysis. These controls ensure the spent fuel pool

configuration and other applicable conditions evaluated in the updated criticality safety analysis remain bounding when compared to current fuel design and plant operating parameters. Specifically, these controls ensure:

1. TS 3.7.15 and TS 4.3 compliance is maintained at all times. Controls are established to ensure that all fuel movement plans into the spent fuel pool are prepared in a manner which ensures continual compliance with the limitations of proposed TS 3.7.15 and TS 4.3, including all intermediate steps during fuel movement.

2. A misloading event beyond the analyzed accident conditions is not credible. Controls are established to ensure that an error in the fuel move planning does not have the potential to result in a misloading accident which is not bounded by the updated criticality safety analysis.

3. Assumptions related to fuel characterization and reactor operation remain valid. Controls are established to ensure that conditions evaluated in the updated criticality safety analysis will remain bounding for both future fuel design changes (pre-irradiation fuel characterization) and future operating conditions (post-irradiation fuel characterization).

Conclusion

In conclusion, the proposed changes to TS 3.7.15 and TS 4.3 allow for continued safe storage of spent fuel at Farley Units 1 and 2.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls.

The requirements for system operability during movement of irradiated fuel are included in the TSs in accordance with 10 CFR 50.36(c)(2), Limiting Conditions for Operation. As required by 10 CFR 50.36(c)(4), design features to be included are 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 categories described in paragraphs (c)(1), (2), and (3) of 10 CFR 50.36. This amendment request concerns 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(4).

At a regulatory level, 10 CFR 50.68(a) requires licensees to select one of two options to satisfy criticality accident requirements: (1) 10 CFR 70.24, or (2) 10 CFR 50.68(b) as highlighted in RIS 2005-05, "NRC Regulatory Issue Summary 2005-05 Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," dated March 23, 2005.

SNC was granted exemption from 10 CFR 70.24 in NRC letter titled "Request for Exemption from 10 CFR 70.24 Criticality Monitoring Requirements – Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC NOS. M95496 and M95497)", dated July 31, 1996.

The critical accident requirements for the current FNP SFP criticality analysis WCAP-14416-P were approved by the NRC in Amendment No 133 to Facility Operating License No. NPF-2 and Amendment No. 125 to Facility Operating License No. NPF-8 for Joseph M. Farley Nuclear Plant, Units 1 and 2 (Accession No. ML013130226). These requirements are also used in the updated Criticality Safety Analysis (Attachment 4/5).

As guidance for reviewing criticality analyses of fuel storage at light-water reactor power plants, the NRC staff issued an internal memorandum on August 19, 1998 (ADAMS Accession No. ML00372B001). This memorandum is known as the "Kopp Letter." The Kopp Letter provides guidance on salient aspects of a criticality analysis. The guidance is germane to boiling-water reactors and pressurized water reactors, and to borated and unborated conditions.

On September 29, 2011, the NRC staff issued the Interim Staff Guidance (ISG) DSS-ISG-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pool," Accession No. ML110620086. The purpose of the ISG is to provide updated review guidance to the NRC staff to address the increased complexity of recent SFP nuclear criticality analyses and operations. The ISG re-baselines the NRC's expectations for spent fuel criticality analysis. The expectations of the ISG were further reinforced in subsequent NRC Information Notice 2011-03, "Nonconservative Criticality Safety Analyses for Fuel Storage," Accession No. ML103090055.

General Design Criterion (GDC) 61 – Fuel storage and handling and radioactivity control, "The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions."

GDC 62 – Prevention of criticality in fuel storage and handling, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes. Preferably by use of geometrically safe conditions.

Additional guidance is available in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," particularly Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, issued March 2007. Section 9.1.1 provides the existing recommendations for performing the review of the nuclear criticality safety analysis of SFPs.

4.2 Precedent

The following license amendment requests and applicable RAIs have been used in the development of the criticality safety analysis and the appropriate sections of the amendment request:

Comanche Peak Units 1 and 2 License Amendment Request for Spent Fuel Pool Technical Specification Changes in a letter dated July 1, 2014, "Comanche Peak Nuclear Power Plant, Units 1 and 2 – Issuance of Amendments RE: Revision to Technical Specifications 3.7.16, "Duel Storage Pool Boron Concentration," 3.7.17, "Spent Fuel Assembly Storage," 4.3, "Fuel Storage," and 5.5, "Programs and Manuals" (TAC NOS. MF1365 and MF1366)," Accession No. ML14160A035.

Palo Verde Units 1, 2, and 3 License Amendment Request for Spent Fuel Pool Technical Specification Changes in a letter dated July 28, 2017, "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendments to Revise Technical Specifications to Incorporate Updated Criticality Safety Analysis (CAC NOS. MF7138, MF7139, and MF7140)," Accession No. ML17188A412.

Prairie Island Units 1 and 2 License Amendment Request for Spent Fuel Pool Criticality Technical Specification Changes in a letter dated November 30, 2017, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendment Revising Spent Fuel Pool Criticality Technical Specification (CAC NOS. MF7121 and MF7122, EPID L-2015-LLA-0002)," Accession No. ML17334A178.

4.3 No Significant Hazards Consideration Determination Analysis

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) 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 amendment was evaluated for impact on the following criticality events and accidents and no impacts were identified: (1) loss of spent fuel pool cooling system, (2) dropping a fuel assembly into an already loaded storage cell, and (3) the misloading of a single fuel assembly or multiple fuel assemblies into a cell for which the restrictions on location, enrichment, or burnup are not satisfied.

   Operation in accordance with the proposed amendment will not change the probability of a loss of spent fuel pool cooling because the changes in the criticality safety analysis have no bearing on the systems, structures, and components involved in initiating such an event. A criticality safety analysis of the limiting fuel loading configuration confirmed that the condition would remain subcritical for a range of normal and accident conditions. The effects of the accident conditions are bounded by the multiple fuel assembly misload accident.

   Operation in accordance with the proposed amendment will not change the probability of a fuel assembly being dropped into an already loaded storage cell because fuel movement will continue to be controlled by approved fuel handling procedures. The consequences of a dropped fuel assembly are not changed; there will continue to be significant separation between the dropped fuel assembly and the active regions of the fuel assemblies. The effects of this accident are bounded by the multiple fuel assembly misload accident.

Operation in accordance with the proposed amendment will not change the probability of a fuel assembly misloading because fuel movement will continue to be controlled by approved fuel selection and fuel handling procedures. These procedures continue to require identification of the initial and target locations for each fuel assembly and fuel assembly insert that is moved. The consequences of a fuel misloading event are not changed because the reactivity analysis demonstrates that the same subcriticality criteria and requirements continue to be met for the multiple fuel assembly misload accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of a criticality 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 potential for criticality in the spent fuel pool is not a new or different type of accident. Storage configurations allowed by Technical Specifications 3.7.15 and 4.3 have been analyzed to demonstrate that the pool remains subcritical.

   The new criticality safety analysis includes analysis of a multiple misload accident scenario; only single misload events were previously analyzed. The inclusion of this analysis does not imply the creation of the possibility of a new accident, but simply expands the boundaries of the analyzed accident conditions to ensure that all potential accidents are properly considered.

   There is no significant change in plant configuration, equipment design or usage of plant equipment. The updated criticality safety analysis assures that the pool will continue to remain subcritical.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change was evaluated for its effect on current margins of safety as they relate to criticality. The margin of safety for subcriticality required by Amendment No. 133 to Facility Operating License No. NPF-2 and Amendment No. 125 to Facility Operating License No. NPF-8 for Joseph M. Farley Nuclear Plant, Units 1 and 2 (Accession No. ML013130226) is unchanged. The updated criticality safety analysis confirms that operation in accordance with the proposed amendment continues to meet the required subcriticality margin.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, SNC 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.

4.4 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.

5. ENVIRONMENTAL CONSIDERATION

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 a 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.

6. REFERENCES

6.1. NEI 12-16, Rev. 3, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants", March 2018.
6.2. K.Wood, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," DSS-ISG-2010-001, Accession Number ML102220567, Nuclear Regulatory Commission, Rockville, MD, August 2010.

ENCLOSURE

Attachment 1

Existing Technical Specification Page Markups

Pages: 3.7.15-1
3.7.15-2
4.0-2
4.0-5
4.0-6
4.0-7
4.0-8
4.0-9
4.0-10
Insert 1 for 4.0-5 (7 Pages)

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in the spent fuel storage pool shall be ~~within the Acceptable Burnup Domain of Figure 3.7.15-1 or~~ in accordance with Specification 4.3.1.1.

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

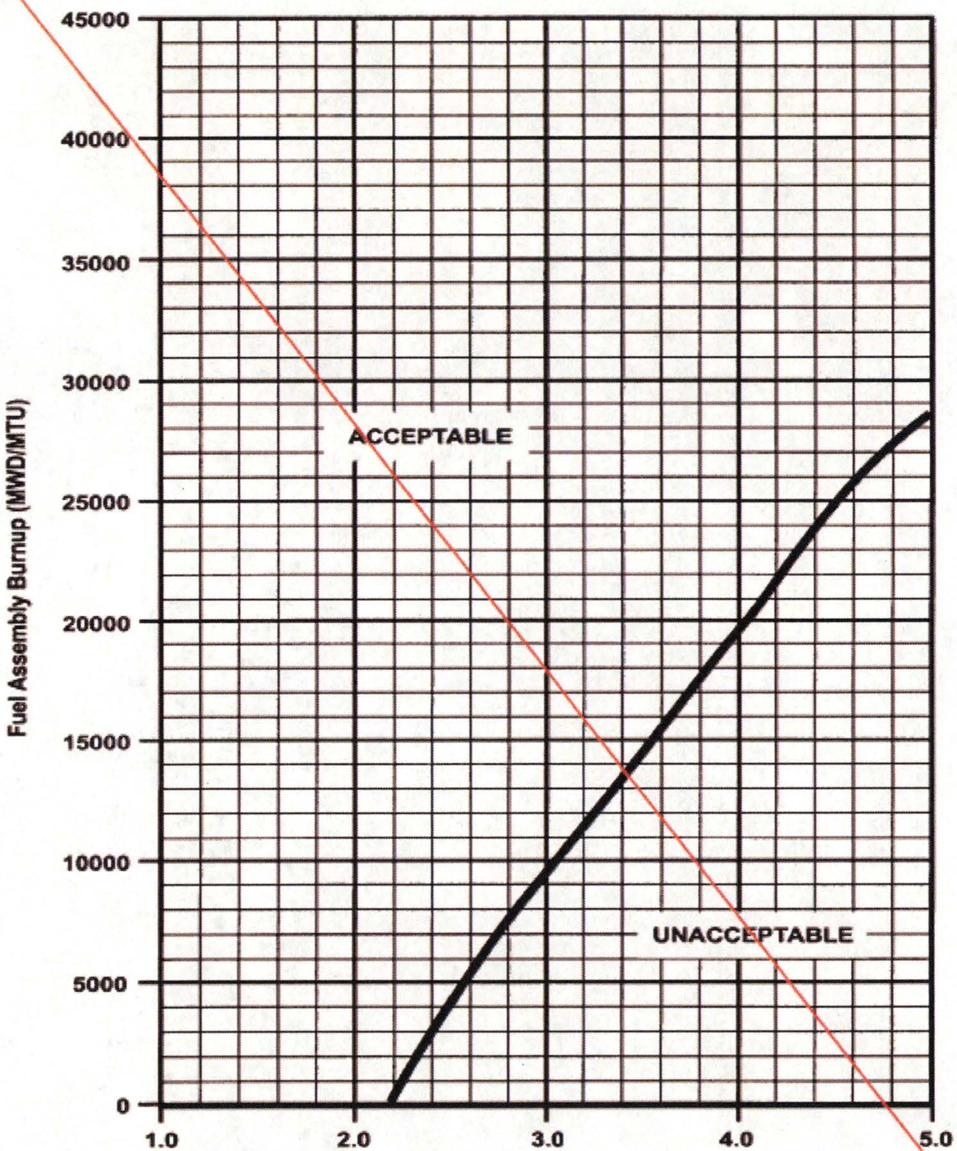
ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure 3.7.15-1 or~~ Specification 4.3.1.1.	Within 7 days following the relocation or addition of fuel assemblies to the spent fuel storage pool.

Delete Figure 3.7.15-1.



Initial U-235 Enrichment (nominal w/o)

Figure 3.7.15-1 Fuel Assembly Burnup Limit Requirements For All Cell Storage

4.0 DESIGN FEATURES

4.3.1.1 (continued)

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3.2.7.2 of the FSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 400 ppm, which includes an allowance for uncertainties and biases as described in Section 4.3.2.7.2 of the FSAR;

d. A nominal 10.75 inch center to center distance between fuel assemblies placed in the fuel storage racks;

New or partially spent fuel assemblies that must be stored according to their combination of discharge burnup and nominal enrichment, decay time since operation, required Integral Fuel Burnable Absorber (IFBA) (if applicable), and must comply with Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5 (as applicable). Each assembly should be stored in an appropriate storage configuration according to its fuel category as specifically described in Table 4.3-1 and geometry based on Figure 4.3-1;

e. ~~New or partially spent fuel assemblies with a combination of discharge burnup and initial enrichment in the "acceptable range" of Figure 3.7.15-1 may be allowed unrestricted storage in the spent fuel storage rack(s) (also shown as the All Cell Storage configuration in Figure 4.3-2);~~

Fuel assemblies that are stored in accordance with every applicable storage array as shown in Figure 4.3-1 of which they are a part (i.e., one fuel assembly can be part of up to four different storage arrays, each storage array shall be in accordance with Figure 4.3-1); and

f. ~~New or partially spent fuel assemblies with a combination of discharge burnup and initial enrichment in the "unacceptable range" of Figure 3.7.15-1 will be stored in compliance with the NRC approved Figures 4.3-1 through 4.3-5. The high enrichment fuel assemblies shown in the Burned/Fresh Storage configuration in Figure 4.3-2, with maximum nominal enrichments > 3.9 weight percent U-235, shall contain sufficient integral burnable absorbers such that a maximum reference fuel assembly $K_{\infty} \leq 1.455$ at 68°F is maintained; and~~

g. Unit 1 only — Damaged fuel assemblies F02, F05, F06, F15, F17, F18, F19, F20, F30, F31, and F32 shall be stored in accordance with ~~Figure 4.3-6.~~ Figure 4.3-2.

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

a. Fuel assemblies with Standard Fuel Assembly fuel rod diameters having a maximum nominal U-235 enrichment of 4.25 weight percent;

(continued)

Delete Figure 4.3-1 and insert revised Figure 4.3-1 and Tables 4.3-1 through 4.3-5 (Insert 1).

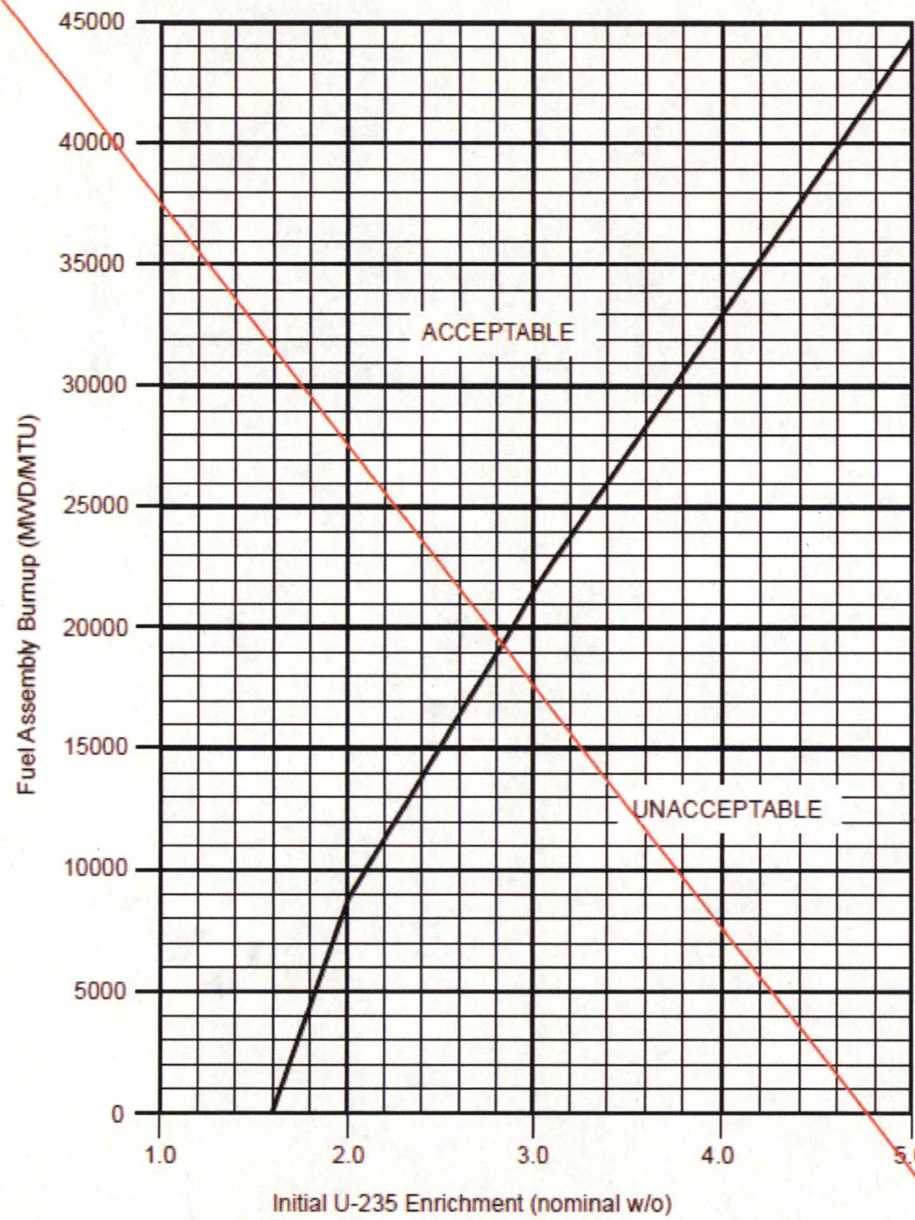
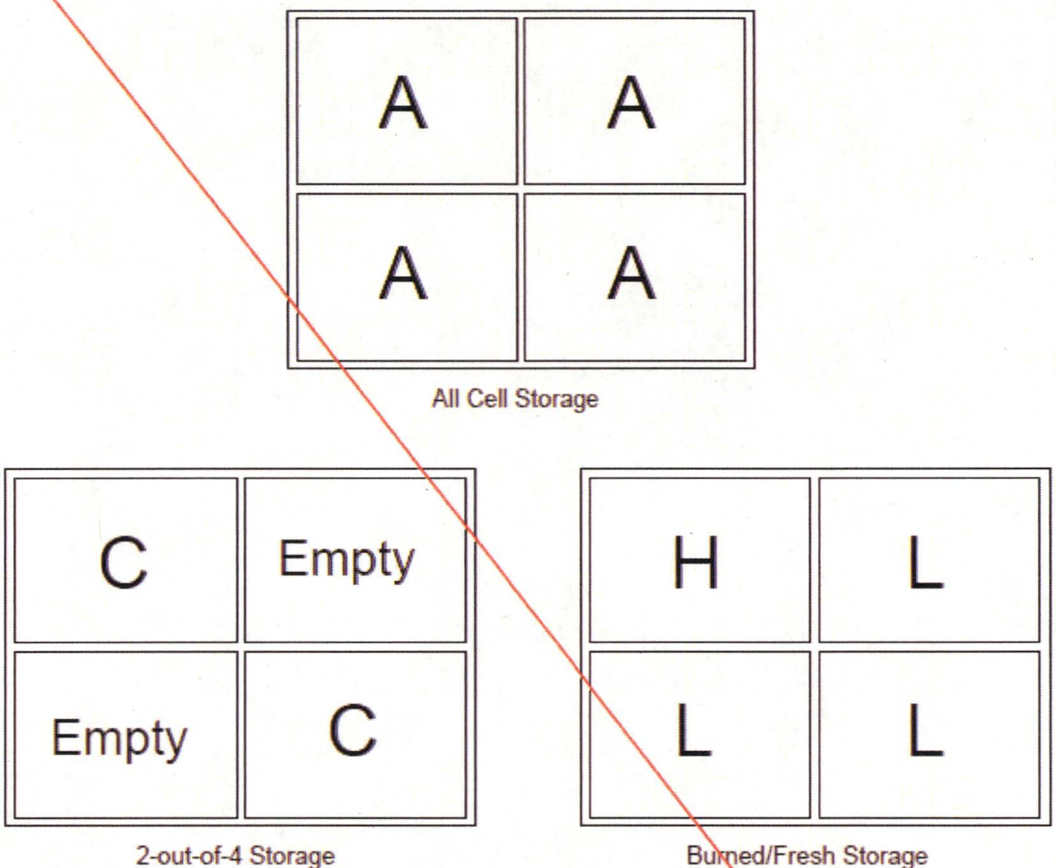


Figure 4.3-1
Fuel Assembly Burnup Limit Requirements for Low Enrichment (L) Assembly of the Burned/Fresh Checkerboard Storage (see Figure 4.3-2)

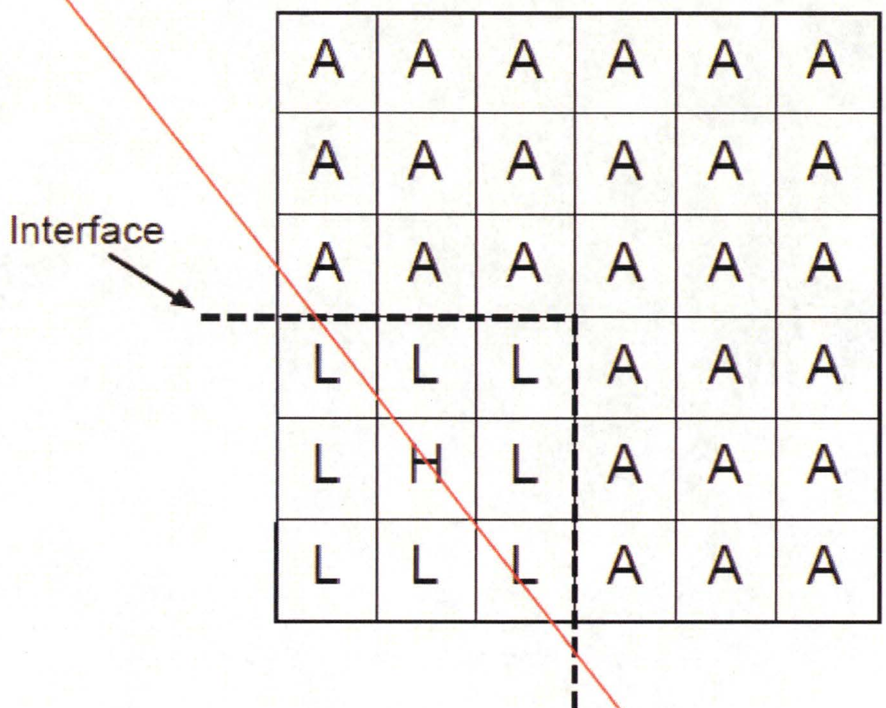
Delete Figure 4.3-2.



Note:
A = All Cell Enrichment (Figure 3.7.15-1)
C = 2-out-of-4 Enrichment (No restriction on enrichment or burnup)
L = Low Enrichment of Burned/Fresh (Figure 4.3-1)
H = High Enrichment of Burned/Fresh (See section 4.3.1.1.f for IFBA requirement)
Empty = Empty Cell

Figure 4.3-2
Spent Fuel Storage Configurations

Delete Figure 4.3-3.



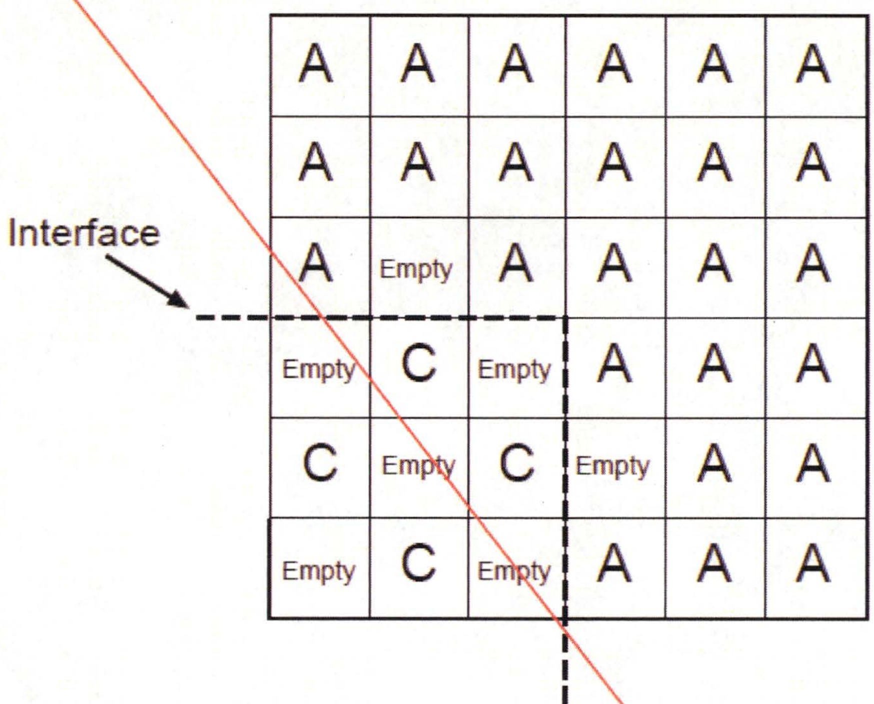
Note:
A = All Cell Enrichment
L = Low Enrichment of Burned/Fresh
H = High Enrichment of Burned/Fresh

Boundary Between All Cell Storage and Burned/Fresh Storage

Note:
1. A row of empty cells can be used at the interface to separate the configurations.
2. It is acceptable to replace an assembly with an empty cell.

Figure 4.3-3
Interface Requirements

Delete Figure 4.3-4.



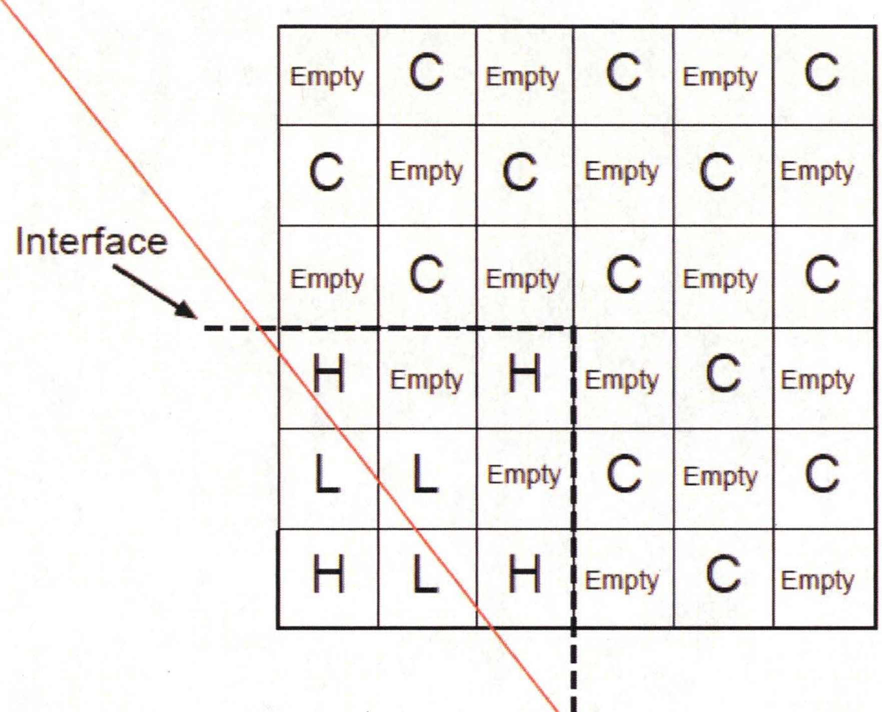
Note:
A = All Cell Enrichment
C = 2-out-of-4 Enrichment
Empty = Empty Cell

Boundary Between All Cell Storage and 2-out-of-4 Storage

Note:
1. A row of empty cells can be used at the interface to separate the configurations.
2. It is acceptable to replace an assembly with an empty cell.

Figure 4.3-4
Interface Requirements

Delete Figure 4.3-5.



Note:
C = 2-out-of-4 Enrichment
L = Low Enrichment of Burned/Fresh
H = High Enrichment of Burned/Fresh
Empty = Empty Cell

Boundary Between 2-out-of-4 Storage and Burned/Fresh Storage

Note:
1. A row of empty cells can be used at the interface to separate the configurations.
2. It is acceptable to replace an assembly with an empty cell.

Figure 4.3-5
Interface Requirements

Renumber Figure 4.3-6 as Figure 4.3-2.

	F31	Empty	F30	F06	
	F18	F17	F19	F02	
	F15	F20	F05	F32	
				Water	

Note: All Assemblies are 3.0 w/o ^{235}U nominal enrichment

4.3-2

Figure ~~4.3-6~~
Damaged Fuel Assembly Configuration
(Unit 1 Only)

Insert 1

Any 2x2 array of storage cells containing fuel shall comply with the requirements of Array A, Array B, or Array C, as applicable.

A. Fuel is divided into two Groups, based on Fuel Type (Standard Fuel Assembly (STD)/Robust Fuel Assembly (RFA) or Optimized Fuel Assembly (OFA)).

B. Arrays A, B and C designate the pattern of fuel which may be stored in any 2x2 Array.

C. Fuel Categories 1-4 are defined in Table 4.3-1.

Array A Two Category 1 assemblies with two empty storage locations The Category 1 fuel assemblies must only be face adjacent to an empty storage location.	1	X
	X	1

Array B One Category 2 assembly with three Category 4 assemblies.	4	4
	4	2

Array C Four Category 3 assemblies.	3	3
	3	3

Notes:

1. Any storage array location designated for a fuel assembly may be replaced with non-fissile material.

2. Empty locations designated with an X must remain completely empty.

Figure 4.3-1
Spent Fuel Pool Loading Restrictions

Insert 1
(Continued)

Notes Continued:

3. Other Fuel Categories are determined as follows:

    a. For STD/RFA assemblies, determine the fitting coefficients $A_1 - A_4$ using Table 4.3-3.

    b. For OFA assemblies, determine the fitting coefficients $A_1 - A_4$ using Table 4.3-4.

    c. For assemblies with Initial Enrichment (En) values greater than or equal to the values in Table 4.3-2, the required Minimum Burnup value (in MWd/MTU) for each Fuel Category is calculated based on initial enrichment, decay time, and the appropriate fitting coefficients. If the fuel assembly burnup is greater than the calculated Minimum Burnup value, then the fuel may be classified into this Fuel Category.

       The equation for Minimum Burnup is:
       - Minimum Burnup (MWd/MTU) = $1{,}000 \times [A_1 \times En^3 + A_2 \times En^2 + A_3 \times En + A_4]$
       - Note: If the computed Minimum Burnup value is negative, zero shall be used.

       The equation for Minimum IFBA required for Fuel Category 2 assemblies as a function of enrichment between 3.2 and 5.0 weight percent Uranium-235 is:
       - Minimum IFBA (rods) = $A_1 \times En^2 + A_2 \times En + A_3$
       - Note: The Minimum IFBA should be rounded up to the next whole number.
       - Note: Below 3.2 weight percent U-235, IFBA is not required.

    d. Decay time is measured in years. For decay times between the values in Tables 4.3-3 and 4.3-4, linear interpolation or the lower decay time value may be used. If interpolation is used, linear interpolation based on actual decay time should be performed between calculated values of Minimum Burnup associated with tabulated Decay Times greater and less than the actual Decay Time. No extrapolation beyond 20 years is permitted.

    e. Initial enrichment (En) is the nominal U-235 enrichment of the central zone region of fuel, excluding axial blankets. If the fuel assembly contains axial regions with different U-235 enrichment values, such as axial blankets, the maximum enrichment value should be utilized. If the computed Minimum Burnup value is negative, zero shall be used.

Figure 4.3-1
Spent Fuel Pool Loading Restrictions
Page 2 of 3

Insert 1
(Continued)

Notes Continued:

4. An empty (water-filled) cell may be substituted for any fuel-containing cell in all storage arrays.

5. Fuel Category 2 fuel which has been operated must have at least 10,000 MWd/MTU of burnup.

Figure 4.3-1
Spent Fuel Pool Loading Restrictions
Page 3 of 3

Insert 1
(Continued)

Table 4.3-1
Fuel Categories Ranked by Reactivity

Fuel Category 1	High Reactivity
Fuel Category 2	
Fuel Category 3	
Fuel Category 4	Low Reactivity

Notes:
1. Assembly storage is controlled through the storage arrays defined in Figure 4.3-1.
2. Fuel Categories are ranked in order of decreasing reactivity, e.g., Fuel Category 2 is less reactive than Fuel Category 1, etc.
3. Each storage cell in an array can only be populated with assemblies of the fuel category defined in the array definition or a lower reactivity fuel category.
4. Fuel Category 1 contains fuel with an initial maximum enrichment up to 5 weight percent U-235. Neither burnup nor IFBA is required.
5. Fuel Category 2 contains fuel with an initial maximum enrichment up to 5 weight percent U-235. Storage of fresh fuel is determined from the minimum IFBA equation and coefficients provided in Table 4.3-5. To be eligible for Fuel Category 2, fuel which has been operated in the reactor requires at least 10,000 MWd/MTU of burnup.
6. Fuel Categories 3 and 4 are determined from the minimum burnup equation and coefficients provided in Table 4.3-3 for STD/RFA fuel and in Table 4.3-4 for OFA fuel.

Table 4.3-2
Maximum Enrichment allowed with 0.0 MWd/MTU Burnup

Fuel Category	RFA/STD	OFA
1	5.0	5.0
2	5.0[1]	5.0[1]
3	2.15	2.15
4	1.70	1.75

Notes:
1. Requires IFBA credit for greater than 3.2 weight percent U-235.
2. For assemblies with an Initial Enrichment below the values listed above, no burnup is required

Insert 1
(Continued)

Table 4.3-3

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for STD/RFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.2251	-2.5199	21.4065	-36.6115
	5	0.3002	-3.4376	24.0978	-38.9002
	10	0.1856	-2.3309	20.2704	-34 6503
	15	0.0892	-1.3905	17.0683	-31.1550
	20	0.0388	-0.9253	15.5082	-29.4500
4	0	-0.6112	4.6655	6.7127	-21.8911
	5	-0.3326	2.0713	12.8468	-26.1880
	10	-0.1305	0.0505	18.3242	-30.7080
	15	0.1360	-2.6856	26.5239	-38.3300
	20	0.2321	-3.7177	29.5977	-41.1200

Insert 1
(Continued)

Table 4.3-4

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for OFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.1692	-1.8852	18.5219	-32.7830
	5	0.0191	-0.4154	13.4482	-27.1777
	10	-0.0705	0.4300	10.5987	-24.0722
	15	-0.1420	1.1146	8 2825	-21.5440
	20	-0.1959	1.6375	6.5093	-19.6130
4	0	0.4957	-6.0715	37.2851	-49.1282
	5	0.7476	-8.7581	45.3241	-56.5172
	10	0.9041	-10.4334	50.3246	-61.0800
	15	1.0799	-12 2326	55.7508	-66.1820
	20	1.2541	-13.9154	60.5977	-70.5720

Insert 1
(Continued)

Table 4.3-5

Fuel Category 2 Coefficients to Calculate the Minimum IFBA Required as a Function of IFBA Thickness and Fuel Type

Fuel Type	IFBA Thickness	Coefficients		
		A_1	A_2	A_3
STD/RFA	1.00X	5.2750	8.3325	-79.9546
	1.25X	3.7476	10.8046	-72.0974
	1.50X	1.8593	19.8050	-81.5075
OFA	1.00X	6.2658	0.8890	-65.4949
	1.25X	3.9144	9.3963	-68.9414
	1.50X	1.5898	21.8436	-84.9630

ENCLOSURE

Attachment 2

Retyped Technical Specification Pages

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in the spent fuel storage pool shall be in accordance with Specification 4.3.1.1.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Specification 4.3.1.1.	Within 7 days following the relocation or addition of fuel assemblies to the spent fuel storage pool.

4.0 DESIGN FEATURES

4.3.1.1 (continued)

b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3.2.7.2 of the FSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 400 ppm, which includes an allowance for uncertainties and biases as described in Section 4.3.2.7.2 of the FSAR;

d. A nominal 10.75 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or partially spent fuel assemblies that must be stored according to their combination of discharge burnup and nominal enrichment, decay time since operation, required Integral Fuel Burnable Absorber (IFBA) (if applicable), and must comply with Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5 (as applicable). Each assembly should be stored in an appropriate storage configuration according to its fuel category as specifically described in Table 4.3-1 and geometry based on Figure 4.3-1;

f. Fuel assemblies that are stored in accordance with every applicable storage array as shown in Figure 4.3-1 of which they are a part (i.e., one fuel assembly can be part of up to four different storage arrays, each storage array shall be in accordance with Figure 4.3-1); and

g. Unit 1 only — Damaged fuel assemblies F02, F05, F06, F15, F17, F18, F19, F20, F30, F31, and F32 shall be stored in accordance with Figure 4.3-2.

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

a. Fuel assemblies with Standard Fuel Assembly fuel rod diameters having a maximum nominal U-235 enrichment of 4.25 weight percent;

(continued)

Any 2x2 array of storage cells containing fuel shall comply with the requirements of Array A, Array B, or Array C, as applicable.

A. Fuel is divided into two Groups, based on Fuel Type (Standard Fuel Assembly (STD)/Robust Fuel Assembly (RFA) or Optimized Fuel Assembly (OFA)).

B. Arrays A, B and C designate the pattern of fuel which may be stored in any 2x2 Array.

C. Fuel Categories 1-4 are defined in Table 4.3-1.

<u>Array A</u> Two Category 1 assemblies with two empty storage locations. The Category 1 fuel assemblies must only be face adjacent to an empty storage location.	1	X
	X	1

<u>Array B</u> One Category 2 assembly with three Category 4 assemblies.	4	4
	4	2

<u>Array C</u> Four Category 3 assemblies.	3	3
	3	3

Notes:

1. Any storage array location designated for a fuel assembly may be replaced with non-fissile material.

2. Empty locations designated with an X must remain completely empty.

Figure 4.3-1
Spent Fuel Pool Loading Restrictions
Page 1 of 3

Notes Continued:

3. Other Fuel Categories are determined as follows:

   a. For STD/RFA assemblies, determine the fitting coefficients $A_1 - A_4$ using Table 4.3-3.

   b. For OFA assemblies, determine the fitting coefficients $A_1 - A_4$ using Table 4.3-4.

   c. For assemblies with Initial Enrichment (En) values greater than or equal to the values in Table 4.3-2, the required Minimum Burnup value (in MWd/MTU) for each Fuel Category is calculated based on initial enrichment, decay time, and the appropriate fitting coefficients. If the fuel assembly burnup is greater than the calculated Minimum Burnup value, then the fuel may be classified into this Fuel Category.

      The equation for Minimum Burnup is:
      - Minimum Burnup (MWd/MTU) = $1{,}000 \times [A_1 \times En^3 + A_2 \times En^2 + A_3 \times En + A_4]$
      - Note: If the computed Minimum Burnup value is negative, zero shall be used.

      The equation for Minimum IFBA required for Fuel Category 2 assemblies as a function of enrichment between 3.2 and 5.0 weight percent Uranium-235 is:
      - Minimum IFBA (rods) = $A_1 \times En^2 + A_2 \times En + A_3$
      - Note: The Minimum IFBA should be rounded up to the next whole number.
      - Note: Below 3.2 weight percent U-235, IFBA is not required.

   d. Decay time is measured in years. For decay times between the values in Tables 4.3-3 and 4.3-4, linear interpolation or the lower decay time value may be used. If interpolation is used, linear interpolation based on actual decay time should be performed between calculated values of Minimum Burnup associated with tabulated Decay Times greater and less than the actual Decay Time. No extrapolation beyond 20 years is permitted.

   e. Initial enrichment (En) is the nominal U-235 enrichment of the central zone region of fuel, excluding axial blankets. If the fuel assembly contains axial regions with different U-235 enrichment values, such as axial blankets, the maximum enrichment value should be utilized. If the computed Minimum Burnup value is negative, zero shall be used.

Figure 4.3-1
Spent Fuel Pool Loading Restrictions
Page 2 of 3

Notes Continued:

4. An empty (water-filled) cell may be substituted for any fuel-containing cell in all storage arrays.

5. Fuel Category 2 fuel which has been operated must have at least 10,000 MWd/MTU of burnup.

Figure 4.3-1
Spent Fuel Pool Loading Restrictions
Page 3 of 3

Table 4.3-1
Fuel Categories Ranked by Reactivity

Fuel Category 1	High Reactivity
Fuel Category 2	
Fuel Category 3	
Fuel Category 4	Low Reactivity

Notes:

1. Assembly storage is controlled through the storage arrays defined in Figure 4.3-1.
2. Fuel Categories are ranked in order of decreasing reactivity, e.g., Fuel Category 2 is less reactive than Fuel Category 1, etc.
3. Each storage cell in an array can only be populated with assemblies of the fuel category defined in the array definition or a lower reactivity fuel category.
4. Fuel Category 1 contains fuel with an initial maximum enrichment up to 5 weight percent U-235. Neither burnup nor IFBA is required.
5. Fuel Category 2 contains fuel with an initial maximum enrichment up to 5 weight percent U-235. Storage of fresh fuel is determined from the minimum IFBA equation and coefficients provided in Table 4.3-5. To be eligible for Fuel Category 2, fuel which has been operated in the reactor requires at least 10,000 MWd/MTU of burnup.
6. Fuel Categories 3 and 4 are determined from the minimum burnup equation and coefficients provided in Table 4.3-3 for STD/RFA fuel and in Table 4.3-4 for OFA fuel.

Table 4.3-2
Maximum Enrichment allowed with 0.0 MWd/MTU Burnup

Fuel Category	RFA/STD	OFA
1	5.0	5.0
2	5.0[1]	5.0[1]
3	2.15	2.15
4	1.70	1.75

Notes:

1. Requires IFBA credit for greater than 3.2 weight percent U-235.
2. For assemblies with an Initial Enrichment below the values listed above, no burnup is required

Table 4.3-3

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for STD/RFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.2251	-2.5199	21.4065	-36.6115
	5	0.3002	-3.4376	24.0978	-38.9002
	10	0.1856	-2.3309	20.2704	-34.6503
	15	0.0892	-1.3905	17.0683	-31.1550
	20	0.0388	-0.9253	15.5082	-29.4500
4	0	-0.6112	4.6655	6.7127	-21.8911
	5	-0.3326	2.0713	12.8468	-26.1880
	10	-0.1305	0.0505	18.3242	-30.7080
	15	0.1360	-2.6856	26.5239	-38.3300
	20	0.2321	-3.7177	29.5977	-41.1200

Table 4.3-4

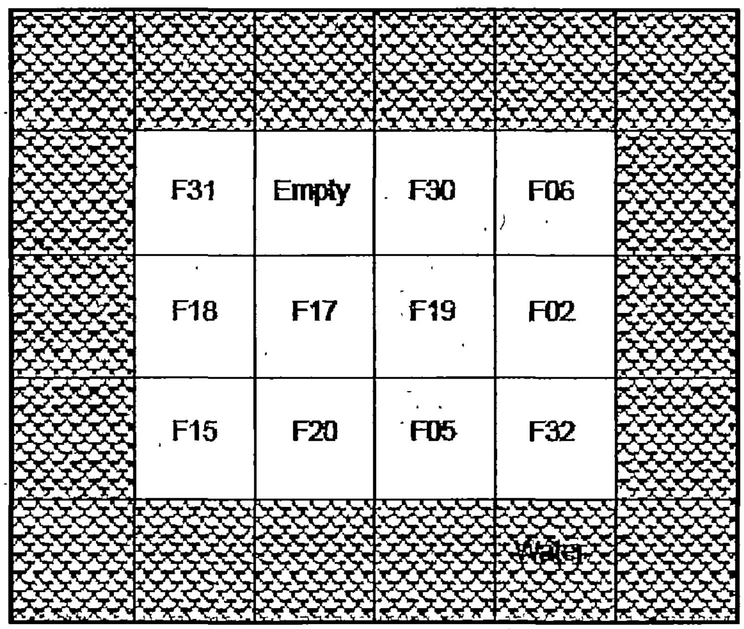
Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for OFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.1692	-1.8852	18.5219	-32.7830
	5	0.0191	-0.4154	13.4482	-27.1777
	10	-0.0705	0.4300	10.5987	-24.0722
	15	-0.1420	1.1146	8.2825	-21.5440
	20	-0.1959	1.6375	6.5093	-19.6130
4	0	0.4957	-6.0715	37.2851	-49.1282
	5	0.7476	-8.7581	45.3241	-56.5172
	10	0.9041	-10.4334	50.3246	-61.0800
	15	1.0799	-12.2326	55.7508	-66.1820
	20	1.2541	-13.9154	60.5977	-70.5720

Table 4.3-5

Fuel Category 2 Coefficients to Calculate the Minimum IFBA Required as a Function of IFBA Thickness and Fuel Type

Fuel Type	IFBA Thickness	Coefficients		
		A_1	A_2	A_3
STD/RFA	1.00X	5.2750	8.3325	-79.9546
	1.25X	3.7476	10.8046	-72.0974
	1.50X	1.8593	19.8050	-81.5075
OFA	1.00X	6.2658	0.8890	-65.4949
	1.25X	3.9144	9.3963	-68.9414
	1.50X	1.5898	21.8436	-84.9630



Note: All Assemblies are 3.0 w/o ^{235}U nominal enrichment

Figure 4.3-2
Damaged Fuel Assembly Configuration
(Unit 1 Only)

ENCLOSURE

Attachment 3

Existing Technical Specification Bases Page Markups

Pages:

B 3.7.14-1
B 3.7.14-2
B 3.7.14-3
B 3.7.14-4
B 3.7.15-1
B 3.7.15-2
B 3.7.15-3
B 3.7.15-4

B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

Fuel assemblies are stored in high density racks. The spent fuel storage racks contain storage locations for 1407 fuel assemblies. Westinghouse 17X17 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 can be stored in any location in the spent fuel storage pool provided the fuel burnup-enrichment combinations ~~are within the limits specified in Figure 3.7.15-1 of the Technical Specifications. Fuel assemblies that do not meet the burnup-enrichment combination of Figure 3.7.15-1 may be stored in the spent fuel storage pool in accordance with the patterns described in Figures 4.3.1-1 through 4.3.1-5.~~ The acceptable storage configurations are based on the ~~"Westinghouse Spent Fuel Rack Criticality Analysis Methodology", WCAP-14416-NP-A, Rev. 1, (Ref. 4) as implemented in the "Farley Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," CAA-97-138, Rev. 1 (Ref. 7).~~

, decay time since operation, and required Integral Fuel Burnable Absorber (IFBA) (if applicable) comply with Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5 (as applicable) of the Technical Specifications.

"J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," WCAP-18414-NP, Rev. 0 (Ref. 4).

This methodology ensures that the spent fuel pool storage rack multiplication factor, K_{eff}, is less than or equal to 0.95, as recommended by ANSI 57.2-1983 (Ref. 3) and NRC Guidance (Refs. 1, 2, and 6). A storage configuration is defined using K_{eff} calculations to ensure that K_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain K_{eff} less than or equal to 0.95. A spent fuel pool boron concentration of 400 ppm will ensure that K_{eff} will be less than or equal to 0.95 for all analyzed combinations of storage patterns, enrichments, and burnups. The treatment of reactivity equivalencing uncertainties, as well as the calculation of postulated accidents crediting soluble boron is described in Ref.4.

Figure 4.3-1 and Table 4.3-1.

The above methodology was used to evaluate storage of Westinghouse 17X17 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 in the FNP spent fuel storage pool. The resulting enrichment and burnup limits are shown in ~~Figure 3.7.15-1.~~ Checkerboard loading patterns are defined to allow storage of fuel assemblies that are not within the acceptable burnup domain of ~~Figure 3.7.15-1.~~ These storage requirements are ~~shown in~~ Technical Specification ~~Figures 4.3.1-1 through 4.3.1-5.~~ A

Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5.

Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5.

(continued)

BASES

BACKGROUND (continued)

spent fuel pool boron concentration of 2000 ppm ensures that no credible boron dilution event will result in a K_{eff} greater than 0.95.

Eleven damaged Westinghouse 17X17 fuel assemblies can be stored in the Unit 1 spent fuel storage pool in the 12 storage cell configuration shown in Technical Specification ~~Figure 4.3.1-6.~~ Figure 4.3-2. The 11 fuel assemblies contain a nominal enrichment of 3.0 weight percent U-235.

APPLICABLE SAFETY ANALYSES

Three accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. The three postulated accidents include a loss of the spent fuel pool cooling system, dropping a fuel assembly into an already loaded storage cell, ~~and~~ the misloading of ~~a~~ ~~fuel assembly~~ single fuel assembly or multiple fuel assemblies into a cell or cells for which the restrictions ~~on~~ location, enrichment, or burnup are not satisfied.

An increase in the temperature of the water passing through the stored fuel assemblies causes a decrease in water density which would normally result in an addition of negative reactivity. However, since Boraflex is not considered to be present in the criticality analysis, and the spent fuel pool water contains a high concentration of boron, a density decrease results in a positive reactivity addition. The effect of an increase in reactivity due to an increase in temperature is bounded by the multiple misload accident.

In the case of a fuel assembly dropped into an already loaded storage cell, the upward axial leakage of that cell will be reduced. However, the overall effect on the storage rack activity would be insignificant, since only the upward axial leakage of a single cell is minimized. In addition, the neutronic coupling between the dropped fuel assembly and the already loaded assembly will be low due to a several inch separation of the active fuel regions due to the fuel assembly bottom nozzle. The effects of this accident are also bounded by the multiple misload accident.

The fuel assembly multiple misloading accident involves the placement of ~~a~~ more than one fresh 5 w/o U-235, unpoisoned fuel assembly into ~~a storage location~~ storage locations for which the requirements on location, enrichment, or burnup are not met. This misload would result in a positive reactivity addition increasing K_{eff} toward 0.95. The amount of soluble boron required to compensate for the positive reactivity added is ~~850 ppm~~ 1710 ppm, which is well below the LCO limit of 2000 ppm.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	A spent fuel pool boron dilution evaluation determined that the volume of water necessary to dilute the spent fuel pool from the LCO limit of 2000 ppm to 400 ppm (the boron concentration required to maintain K_{eff} less than or equal to 0.95) is approximately 480,000 gallons. A spent fuel pool dilution of this volume is not a credible event, since it would require this large volume of water to be transferred from a source to the spent fuel pool, ultimately overflowing the pool. This event would be detected and terminated by plant personnel prior to exceeding a K_{eff} of 0.95. The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The fuel storage pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 5. The specified boron concentration of 2000 ppm ensures that the spent fuel pool K_{eff} will remain less than or equal to 0.95 due to a postulated fuel assembly multiple misload accident (~~850 ppm~~ 1710 ppm) or boron dilution event (400 ppm).
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.
ACTIONS	A.1 and A.2 The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. When the concentration of boron in the fuel storage 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 fuel assemblies. Action is also initiated to restore the concentration of boron simultaneously with suspending movement of fuel assemblies.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, June, 1987.

2. USNRC Spent Fuel Storage Facility Design Bases (for Comment) Proposed Revision 2, 1981.

3. ANS, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," ANSI/ANS-57.2-1983.

WCAP-18414-NP, Rev. 0, "J.M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," September, 2019.

4. ~~WCAP-14416-NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November, 1996.~~

5. FSAR, Section 4.3.2.7.2.

6. NRC, Letter to all Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

~~7. "Farley Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," CAA 97 138, Rev. 1.~~

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

Fuel assemblies are stored in high density racks. The spent fuel storage racks contain storage locations for 1407 fuel assemblies. Westinghouse 17X17 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 can be stored in any location in the spent fuel storage pool provided the fuel burnup-enrichment combinations , decay time since operation, and required Integral Fuel Burnable Absorber (IFBA) (if applicable) comply with Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5 (as applicable) of the Technical Specifications. ~~are within the limits specified in Figure 3.7.15-1 of the Technical Specifications. Fuel assemblies that do not meet the burnup-enrichment combination of Figure 3.7.15-1 may be stored in the spent fuel storage pool in accordance with the patterns described in Figures 4.3.1-1 through 4.3.1-5.~~ The acceptable storage configurations are based on the "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," WCAP-18414-NP, Rev. 0 (Ref. 1). ~~"Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, Rev. 1, (Ref. 1) as implemented in "Farley Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," CAA-97-138, Rev. 1 (Ref. 2).~~

The following storage configurations and enrichment limits were evaluated in the spent fuel rack criticality analysis:

Westinghouse 17X17 fuel assemblies are stored according to their fuel type (Standard Fuel Assembly (STD)/ Robust Fuel Assembly (RFA) or Optimized Fuel Assembly (OFA)), decay time since last operation, nominal enrichment and burnup. Westinghouse 17X17 fuel assemblies with nominal enrichments less than or equal to 5.0 weight percent U-235 can be stored in Array A as Fuel Category 1 as shown in Figure 4.3-1. In the Array A checkerboard storage arrangement, 2 fuel assemblies can be stored corner adjacent with empty storage cells.

~~Westinghouse 17X17 fuel assemblies with nominal enrichments less than or equal to 2.15 weight percent U-235 can be stored in any cell location as shown if Figure 4.3.1-2. Fuel assemblies with initial nominal enrichments greater than these limits must satisfy a minimum burnup requirement as shown in Figure 3.7.15-1.~~

Westinghouse 17X17 fuel assemblies can be stored in a burned/fresh arrangement (Array B, Fuel Categories 4 and 2) of a 2X2 matrix of storage cells as shown in Figure 4.3-1. In the Array B arrangement, assemblies must satisfy the minimum burnup and enrichment requirements for Fuel Category 4 assemblies as shown in Figure 4.3-1, and Tables 4.3-1, 4.3-3, and 4.3-4, or Fuel Category 2 assemblies must meet the enrichment and IFBA requirements of Figure 4.3-1 and Tables 4.3-1 and 4.3-5.

~~Westinghouse 17X17 fuel assemblies with nominal enrichments less than or equal to 5.0 weight percent U-235 can be stored in a 2 out of 4 checkerboard arrangement as shown in Figure 4.3.1-2. In the 2 out of 4 checkerboard storage arrangement, 2 fuel assemblies can be stored corner adjacent with empty storage cells.~~

~~Westinghouse 17X17 fuel assemblies can be stored in a burned/fresh checkerboard arrangement of a 2X2 matrix of storage cells as shown in Figure 4.3.1-2. In the burned/fresh 2X2 checkerboard arrangement, three of the fuel assemblies must have an initial nominal enrichment less than or equal to 1.6 weight percent U-235, or satisfy a minimum burnup requirement for higher initial enrichments as shown in Figure 4.3.1-1.~~

Westinghouse 17X17 fuel assemblies can be stored in a uniform "all-cell" arrangement (Array C) of a 2X2 matrix of storage cells as shown in Figure 4.3-1. In the Array C all-cell arrangement, assemblies must satisfy the minimum burnup and enrichment requirements of Fuel Category 3 assemblies as shown in Figure 4.3-1 and Tables 4.3-1, 4.3-3, and 4.3-4.

BASES

BACKGROUND (continued)

~~The fourth fuel assembly must have an initial nominal enrichment less than or equal to 3.9 weight percent U-235, or satisfy a minimum Integral Fuel Burnable Absorber requirement for higher initial enrichments to maintain the reference fuel assembly K_∞ less than or equal to 1.455 at 68°F.~~

4.3-2

Eleven damaged Westinghouse 17X17 fuel assemblies can be stored in the Unit 1 spent fuel storage pool in a 12 storage cell configuration surrounded by empty cells as shown in Technical Specification Figure ~~4.3.1-6~~. The 11 fuel assemblies contain a nominal enrichment of 3.0 weight percent U-235.

APPLICABLE SAFETY ANALYSES

or cells

Three accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. The three postulated accidents include a loss of the spent fuel pool cooling system, dropping a fuel assembly into an already loaded storage cell, and the misloading of ~~a fuel assembly~~ into a cell for which the restrictions ~~on location~~, enrichment, or burnup are not satisfied.

single fuel assembly or multiple fuel assemblies

An increase in the temperature of the water passing through the stored fuel assemblies causes a decrease in water density which would normally result in an addition of negative reactivity. However, since Boraflex is not considered to be present in the criticality analysis, and the spent fuel pool water contains a high concentration of boron, a density decrease results in a positive reactivity addition. The effect of an increase in reactivity due to an increase in temperature is bounded by the misload accident.

multiple

In the case of a fuel assembly dropped into an already loaded storage cell, the upward axial leakage of that cell will be reduced. However, the overall effect on the storage rack activity would be insignificant, since only the upward axial leakage of a single cell is minimized. In addition, the neutronic coupling between the dropped fuel assembly and the already loaded assembly will be low due to a several inch separation of the active fuel regions due to the fuel assembly bottom nozzle. The effects of this accident are also bounded by the misload accident.

multiple

(continued)

more than one fresh 5 w/o U-235, unpoisoned

storage locations

BASES

multiple

APPLICABLE SAFETY ANALYSES (continued)

The fuel assembly misloading accident involves the placement of ~~a~~ fuel assembly into ~~a storage location~~ for which the requirements on location, enrichment, or burnup are not met. This misload would result in a positive reactivity addition increasing K_{eff} toward 0.95. The amount of soluble boron required to compensate for the positive reactivity added is ~~850 ppm~~, which is well below the LCO limit of 2000 ppm.

1710 ppm

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool ensure the K_{eff} of the spent fuel storage pool will always remain < 0.95, assuming the pool to be flooded with borated water. The combination of initial enrichment ~~and burnup are specified in Figure 3.7.15-1 for the All Cell Storage Configuration. Other acceptable enrichment, burnup, and checkerboard storage configurations are specified in Figures 4.3.1-1 through 4.3.1-6.~~

, decay time, IFBA requirements, and/or burnup of the fuel assembly are specified in Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5 for all spent fuel pool storage configurations.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with the acceptable combination of initial enrichments, burnup, and storage configurations, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with ~~Figure 3.7.15-1 or~~ Specification 4.3.1.1.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means (e.g., Core Loading Plan, Tote computer code output or TrackWorks program) that the initial enrichment ~~and burnup of the fuel assembly is within the acceptable burnup domain of Figure 3.7.15-1. For fuel assemblies in the unacceptable range of Figure 3.7.15-1, performance of this SR will also ensure compliance with Specification 4.3.1.1.~~

, decay time, IFBA requirements, and/or burnup of the fuel assembly is within the acceptable burnup domain of Figure 4.3-1, Table 4.3-1, and Tables 4.3-3 through 4.3-5.

The frequency of within 7 days following the relocation or addition of fuel assemblies to the spent fuel storage pool ensures that fuel assemblies are stored within the configuration analyzed in the spent fuel rack criticality analysis. This surveillance would be performed after all of the fuel handling is completed during a refueling outage, or new fuel assemblies are placed into the spent fuel pool.

This SR does not have to be performed following interruptions in fuel handling during defined fuel movements as described above (i.e., it is only required after all fuel movement associated with refueling operations is completed) or if only certain fuel assemblies are relocated to different storage locations within the pool (only the moved assemblies must be verified).

The 7 day allowance for completion of this Surveillance provides adequate time for completion of a spent fuel pool inventory verification while minimizing the time that a fuel assembly could be misloaded during a refueling or the placement of new fuel assemblies into the spent fuel pool. The boron concentration required by Specification 3.7.14 ensures that the spent fuel rack K_{eff} remains within limits until the spent fuel pool inventory verification is performed.

REFERENCES

WCAP-18414-NP, Rev. 0, "J.M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," September, 2019.

1. ~~WCAP-14416-NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November, 1996.~~

~~2. "Farley Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," CAA-97-138, Rev. 1.~~

ENCLOSURE

Attachment 6

Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-19-4943, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Camille T. Zozula, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of WCAP-18414-P, "J. M. Farley Units 1 & 2 Spent Fuel Pool Criticality Safety Analysis" be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

AFFIDAVIT

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters

AFFIDAVIT

refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 12 Sept 2019

Camille T. Zozula, Manager
Fuel Plant & Radioactive Materials

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (5)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE