ENCLOSURE 6 CONTAINS PROPRIETARY INFORMATION – WITHHOLD FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

Xcel Energy®

L-PI-11-075
10 CFR 50.90

August 19, 2011

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

Prairie Island Nuclear Generating Plant, Units 1 and 2
Dockets 50-282 and 50-306
Renewed License Nos. DPR-42 and DPR-60

License Amendment Request for Spent Fuel Pool Criticality Changes

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the renewed operating licenses for Prairie Island Nuclear Generating Plant (PINGP). Specifically, NSPM proposes to revise Technical Specification (TS) 3.7.17, "Spent Fuel Pool Storage" and TS 4.3.1, "Fuel Storage Criticality."

The proposed TS changes will correct non-conservatisms in the Spent Fuel Pool (SFP) criticality analysis-of-record, which have translated into non-conservative TS. These non-conservatisms have been addressed in the PINGP Corrective Action Program, and operability has been restored through interim administrative restrictions on SFP loading patterns that are more restrictive than the TS. The proposed License Amendment Request (LAR) is being submitted to address the need for timely amendments to supersede the current non-conservative TS in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety". The proposed TS changes would allow the use of new Spent Fuel Pool (SFP) loading patterns to maintain subcriticality under all postulated conditions.

Enclosure 1 to this letter provides the evaluation of the proposed TS changes and their supporting justifications, including a no significant hazards determination. Enclosure 2 provides the existing TS pages marked-up to show the proposed changes. Enclosure 3 provides, for information only, the existing TS Bases pages marked-up to show the associated proposed Bases changes. Final TS Bases changes will be implemented pursuant to TS 5.5.12, "Technical Specifications (TS) Bases Control Program," at the time the amendment is implemented. Enclosure 4 provides the non-proprietary version of the Westinghouse Report, WCAP-17400-NP, Revision 0, "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis," dated July 2011. Enclosure 5 provides the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-11-3211, accompanying Affidavit, Proprietary Information Notice, and copyright notice. Enclosure 6 contains the proprietary version of the Westinghouse Report, WCAP-17400-P, Revision 0.

1717 Wakonade Drive East • Welch, Minnesota 55089-9642
Telephone: 651.388.1121

A001
NRR

As Enclosure 6 contains information proprietary to Westinghouse Electric Company LLC, it is supported by the Enclosure 5 affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items provided in Enclosure 6 of this letter or the supporting Westinghouse affidavit should reference CAW-11-3211 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Enclosure 7 provides a list of regulatory commitments made by NSPM in this submittal.

NSPM has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact assessment need not be prepared.

A copy of this submittal, including the Determination of No Significant Hazards Consideration, without Enclosures 2 through 7, is being forwarded to the designated State of Minnesota official pursuant to 10 CFR 50.91(b)(1).

NSPM requests approval of this proposed amendment by February 28, 2013. Once approved, the amendment will be implemented within 120 days.

If there are any questions or if additional information is needed, please contact Glenn Adams at 612-330-6777.

<u>Summary of Commitments</u>

Enclosure 7 provides a list of regulatory commitments made by NSPM in this submittal.

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

Executed on 19 August 2011

Mark A. Schimmel

Mark A. Schimmel
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosures (7)

cc: Regional Administrator, Region III, USNRC
Project Manager, Prairie Island Nuclear Generating Plant, USNRC
Resident Inspector, Prairie Island Nuclear Generating Plant, USNRC
State of Minnesota (without enclosures 2 through 7)

ENCLOSURE 1

Evaluation of the Proposed Change

License Amendment Request for Spent Fuel Pool Criticality Changes

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the renewed operating licenses for Prairie Island Nuclear Generating Plant (PINGP). Specifically, NSPM proposes to revise Technical Specification (TS) 3.7.17, "Spent Fuel Pool Storage" and TS 4.3.1, "Fuel Storage Criticality" to provide new Spent Fuel Pool (SFP) loading restrictions that meet subcriticality criteria for all postulated conditions.

2.0 DETAILED DESCRIPTION

The proposed changes to the TS are as follows:

2.1 Proposed Change to TS 3.7.17, "Spent Fuel Pool Storage"

Currently, spent fuel pool loading restrictions are described in two different sections of the TS. Section 3.7.17 provides the Limiting Condition for Operation (LCO) for SFP loading and provides a figure that must be used to distinguish whether a fuel assembly qualifies for "unrestricted" or "restricted" loading in the SFP. Determining factors in applying loading restrictions are initial enrichment, burnup, and decay time of the fuel assembly. For those assemblies that are "restricted", TS 4.3.1 provides appropriate loading patterns and the associated minimum burnup and decay time requirements that must be met, along with the interface requirements imposed between loading patterns.

The proposed change will replace all previous loading restrictions in TS 3.7.17 and 4.3.1.1 with new loading restrictions. Specific to TS 3.7.17, the proposed change will remove TS Figure 3.7.17-1 and refer to TS 4.3.1.1 for all SFP loading restrictions; thereby eliminating the concept of "restricted" and "unrestricted" loading in the SFP.

In addition, the proposed TS 3.7.17 will expand the LCO and Surveillance Requirement (SR) 3.7.17.1 to address the storage of fuel inserts and hardware.

2.2 Proposed Change to TS 4.3.1, "Fuel Storage Criticality"

The proposed change to TS 4.3.1.1.c will revise the value of SFP boron concentration that ensures all normal conditions in the SFP maintain a neutron multiplication factor (k_{eff}) less than or equal to 0.95.

The proposed change will modify sub-item (e) to apply the new loading requirements of TS Figure 4.3.1-1 and expand the scope to include fuel inserts and hardware. Sub-item (f) is eliminated entirely. Accordingly, the proposed change will replace current loading restrictions in TS Figures 4.3.1-1 through 4.3.1-

4 with new loading restrictions embodied by TS Tables 4.3.1-1 through 4.3.1-3 and TS Figure 4.3.1-1.

The proposed change will also delete TS 4.3.1.3 in its entirety with no replacement. Currently, this TS has imposed a 10 CFR 72 spent fuel cask loading restriction that is more appropriately addressed in Prairie Island Independent Spent Fuel Storage Installation Technical Specifications (Special Nuclear Material license SNM-2506).

The proposed Figure 4.3.1-1 makes provisions for spent fuel pool contents that were not previously included in the TS. These contents include consolidated rod storage canisters, failed fuel baskets, and fuel assembly inserts.

The proposed TS changes described above are shown as mark-ups to the current TS on the pages provided in Enclosure 2.

The TS Bases will also be revised for consistency with the proposed TS changes. A markup of the TS Bases pages reflecting these changes is provided in Enclosure 3 for information only. The proposed TS Bases changes will be implemented in accordance with TS 5.5.12, "Technical Specifications (TS) Bases Control Program," at the same time that the proposed TS changes of this License Amendment Request (LAR) are implemented.

2.3 Other Proposed Changes to the Current Licensing Basis

In addition to the proposed TS amendments discussed above, this LAR also proposes the following changes to the current licensing bases for which NRC approval is requested:

- The proposed amendment would change the regulatory basis for SFP criticality analysis from 10 CFR 70.24 to 10 CFR 50.68(b), which would allow for elimination of criticality accident monitoring requirements while maintaining the subcriticality criteria defined by 10 CFR 50.68(b).
- The proposed amendment would change the evaluation methodology used for SFP criticality analysis to better comport with recent NRC guidance; namely, Draft Interim Staff Guidance (ISG) DSS-ISG-2010-01 (Reference 6.1). The requested evaluation methodology is described in Enclosure 6.

3.0 TECHNICAL EVALUATION

3.1 Design Description

Prairie Island Units 1 and 2 share a common spent fuel pool that employs one modular storage rack design throughout. As described in PINGP Updated Safety Analysis Report (USAR) Section 10.2.1, the storage rack design originally credited Boraflex neutron absorber panels between the storage cells to help meet

subcriticality criteria. These Boraflex panels are degraded and have not been credited in the current design basis. The rack design does benefit from a dedicated "flux-trap" design that provides a minimum gap between cells. Key design parameters for the storage racks are provided in USAR Section 10.2.1 and in the Enclosure 6 analysis.

To ensure stored fuel remains in a subcritical configuration during any normal or accident condition, strict administrative controls require that any fresh (new) fuel assembly or spent fuel assembly loaded into a storage rack is first evaluated to ensure it meets the loading restrictions of TS 3.7.17 and 4.3.1. Currently, each fuel assembly is qualified for a storage location based on several key parameters that include initial enrichment, burnup, decay time, and gadolinia content. Certain parameters (e.g., initial enrichment and gadolinia content) are determined from fuel records. Other parameters (e.g., burnup and decay time) are determined from core operating records. The value of burnup is the average assembly exposure in megawatt days per metric ton uranium (MWD/MTU) and is currently calculated using an industry standard core power distribution system called BEACON™ (Best Estimate Analyzer for Core Operations - Nuclear); however, other suitable methods have been used previously.

Once an assembly is selected for placement based on the required characteristics, procedures ensure that the fuel assembly is qualified for its new location, and that it is safely placed in the designated location.

The spent fuel storage racks are designed so that it is impossible to insert assemblies between rack modules or between rack modules and the spent fuel pool wall. Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack that limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and/or maintaining soluble neutron poison (i.e., boron) in the spent fuel pool water.

The required subcriticality margin of safety for the stored fuel is assured with the soluble boron present in the spent fuel pool. TS 3.7.16 requires a minimum soluble boron concentration of 1800 parts per million (ppm) whenever fuel is present in the spent fuel pool. This boron concentration provides significant margin above the current value (464 ppm[1]) required to maintain an effective neutron multiplication factor (k_{eff}) ≤ 0.95 under normal conditions. Further, this TS value also provides margin above the current value (730 ppm[2]) required to maintain k_{eff} ≤ 0.95 under the limiting accident conditions (fuel misplacement).

Additionally, plant design features and operator responsiveness ensure that the credible spent fuel pool dilution event (initiated at the TS minimum concentration of

---

[1] Spent Fuel Criticality analysis of record determined that 464 ppm of soluble boron was required to maintain the most reactive (i.e., the limiting) normal configuration k_{eff} less than or equal to 0.95.
[2] Spent Fuel Criticality analysis of record determined that 730 ppm of soluble boron was required to maintain the most reactive (i.e., the limiting) accident configuration k_{eff} less than or equal to 0.95.

1800 ppm) will be terminated before the SFP boron concentration reaches 750 ppm. This termination point provides ample margin to the current boron concentration (464 ppm) that ensures the limiting normal configuration stays below k_{eff} 0.95.

Fuel designs employed at PINGP are described in USAR Section 3.1. The original design was Westinghouse 14x14 Standard, and the most recent design in use is the Westinghouse 422 Vantage+ (422V+). However, several variations of 14x14 fuel have been used, including several Exxon designs. In addition to fuel design changes, several core design and operational changes have been made over the plant's operating history that would have a bearing on how the nuclear fuel is depleted during operation. For instance, Burnable Poison Rods (BPRs) were inserted into certain unrodded assembly positions for several cycles as a fixed burnable neutron poison. All applicable design variations and operating variations are evaluated in the Enclosure 6 report.

Another variation in fuel design applicable to the Spent Fuel Criticality Analysis (SFCA) resulted from the fuel consolidation campaign that was conducted in 1987. This consolidation project involved removing the fuel rods from two fuel assemblies and reconfiguring them into a close-packed triangular array; packaged into a specially-design canister. In this manner, 36 assemblies were consolidated into 18 canisters. The project is further described in USAR Section 10.2.1.5.

Other variations on fuel design (failed fuel baskets) and other spent fuel pool materials of interest (e.g., assembly structural materials from the fuel consolidation project) are described further in the SFCA (Enclosures 4 and 6) and supporting calculations.

In August 2010, non-conservatisms were identified in the spent fuel criticality analysis-of-record. Certain plant parameters used in the calculation of nuclear fuel depletion were nonconservative, including the assumed values for operating core fuel temperature, core boron concentration, and the axial burnup profile. These non-conservatisms have been addressed in the PINGP Corrective Action Program, and operability has been restored through implementation of interim administrative restrictions on SFP loading patterns that are more restrictive than Technical Specifications. The proposed LAR is being submitted to address the need for timely amendments to supersede the current non-conservative TS pursuant to NRC Administrative Letter 98-10. The proposed TS changes would allow the use of new SFP loading patterns to maintain subcriticality under all postulated conditions.

With respect to the regulatory requirements of 10 CFR 70.24, the following design and operating description is applicable. In the new fuel pit (NFP) area, a radiation monitor designated R-28 serves to detect any new fuel pool criticality accident that might occur. Also, operators conduct SFP / NFP area evacuation drills in preparation for any possible criticality accident. These measures help satisfy the

plant's regulatory obligation to 10 CFR 70.24, "Criticality Accident Requirements".

The proposed amendments involve no physical modifications to the SFP storage racks or to any other system, structure, or component. No change to the minimum SFP boron concentration limit is required. The only physical effect associated with this proposed amendment will be the re-configuration of fuel in the SFP storage racks.

3.2 Current Licensing Basis

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). Historically and currently, NSPM has chosen to adopt 10 CFR 70.24, Criticality Accident Requirements. However, to demonstrate appropriate margin in its current criticality analyses, NSPM has chosen to apply the criticality acceptance criteria of 10 CFR 50.68(b) without adopting the regulation. These criteria are represented in TS 4.3.1.1 and TS 4.3.1.2.

For the criticality analysis of spent fuel pool abnormal and accident conditions, the current licensing basis uses soluble boron credit and applies the double contingency principle to demonstrate a $k_{eff} \leq 0.95$ for all postulated scenarios. This criterion is described in USAR Section 10.2.1. This $k_{eff} \leq 0.95$ criterion for accidents is more conservative than regulatory guidance which establishes subcriticality ($k_{eff} < 1.0$) as an acceptable limit for accidents.

The USAR describes the applicable PINGP General Design Criterion (GDC-66) as follows: Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. The design and analytical approach to satisfying this criterion is described in USAR Section 10.2.1 and summarized below.

As described in USAR Section 10.2.1 and TS 4.3.1, the New Fuel Pit has been designed and analyzed to maintain $k_{eff} \leq 0.95$ if fully flooded with unborated water, and ≤ 0.98 if accidentally filled with a low-density moderator which resulted in optimum low density moderation conditions. These design criteria are comparable to those described by 10 CFR 50.68(b)(2) and (3) for fresh fuel storage racks.

The design basis for preventing criticality outside the reactor requires a 95% probability at a 95% confidence level that the k_{eff} of the fuel assembly array will be less than 1.0 in the spent fuel pool and less than 0.95 in the new fuel pit with consideration of uncertainties. TS (Design Features 4.3) give special consideration for the "allowance for uncertainties" to be used in criticality analyses. That allowance is defined in the Westinghouse Spent Fuel Rack Criticality Analysis Methodology described in WCAP-16517-NP (for the SFP). Storage configurations have been defined in the TS to ensure that the spent fuel rack k_{eff} will be less than

1.0 including uncertainties and tolerances on a 95/95 basis, without the presence of any soluble boron in the storage pool. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions and to provide subcritical margin such that the spent fuel pool k_{eff} is maintained less than or equal to 0.95. In addition to reactivity credit for soluble boron, both of the k_{eff} calculations (0.95 and 1.0) also take credit for the radioactive decay time of the spent fuel, and the presence of gadolinium absorber, which is mixed in the fuel pellets and is an integral part of some fuel rods.

The Prairie Island spent fuel racks have been analyzed to allow storage of fuel assemblies with nominal enrichments up to 5.0 weight percent (w/o) uranium-235 (U-235) in all storage cell locations using credit for checkerboarding and burnup. The analysis does not take any credit for the presence of the spent fuel rack Boraflex neutron absorber panels which are believed to be in a degraded condition.

Currently, the USAR (Section 10.2.1) describes special fuel configurations that deviate from standard fuel assembly construction. These configurations include the Fuel Rod Storage Canister (FRSC), the Failed Fuel Pin Basket (FFPB), and the Consolidated Rod Storage Canister (CRSC). The design criteria and criticality storage limits for these devices are described in the USAR rather than the TS.

TS 4.3.1.3 establishes a SFP soluble boron concentration requirement that applies when a spent fuel cask is in the SFP. This restriction was added in 1992 by PINGP License Amendments 99/92 (Reference 6.2), at a time when the site-specific Independent Spent Fuel Storage Installation (ISFSI) license was being pursued. Subsequently, in October 1993, the Prairie Island site-specific ISFSI license SNM-2506 (Reference 6.3) was issued to also establish SFP soluble boron requirements during cask loading and unloading operations.

3.3 Justification for the Proposed Changes

3.3.1 Justification for Technical Specification Changes

In a broad sense, the proposed changes to TS would implement loading configurations analyzed to satisfy the same criticality criteria currently in effect at PINGP; the criteria derived from 10 CFR 50.68(b) and supporting regulatory guidance. New conservatisms in the analysis have resulted in increased margins of safety to those regulatory limits. Some of these conservatisms were made while aligning the analysis methods to those prescribed in the new regulatory guidance Draft Interim Staff Guidance DSS-ISG-2010-01, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools (hereafter referred to as the "Draft ISG"). These analysis methods are described in detail in Enclosure 6, which describes how the analysis conservatively selects input parameters and develops analytical methods.

In a specific sense, the proposed restrictions of TS 3.7.17 and 4.3.1 are provided to reflect the new analysis results that revised the fuel categories and loading configurations to meet the regulatory criteria. Specific changes are also made to improve the TS structure and expand the scope to include the placement of fuel inserts and other hardware that may affect criticality.

The proposed revision to TS 3.7.17 eliminates Figure 3.7.17-1 and the related terms of "restricted" and "unrestricted". In effect, the new loading restrictions have made this concept of "unrestricted" storage obsolete because there are no "unrestricted" fuel categories in the proposed TS. Even a fuel assembly that meets the high-burnup Category 6 must meet other restrictions. For example, it must be placed in a qualified array and meet the prescribed interface requirements. Therefore, the concept of "restricted / unrestricted" storage is eliminated and replaced with a complete set of new loading restrictions embodied entirely in TS 4.3.1.1.

The changes in TS 3.7.17 to include fuel inserts and other hardware is conservative and appropriate considering that the proposed SFCA recognizes the potential effect that these elements can have on criticality of the stored fuel. Provisions for the storage of these items are described in TS 4.3.1.1 and are justified based on the supporting SFCA.

TS 4.3.1.1 continues to provide the same criticality criteria used in the current licensing basis except that the value of soluble boron concentration that correlates to the $k_{eff} \leq 0.95$ condition is reduced to correlate with the results of the new SFCA. The minimum boron concentration in the SFP required to maintain k_{eff} for all permissible fuel storage arrangements less than or equal to 0.95 under normal conditions is reduced from 730 ppm to 400 ppm. This 400 ppm value was selected (for the TS) to provide margin to the value of 350 ppm (359 ppm at the reduced boron-10 atom % of 19.4) which was calculated for the limiting normal configuration described in Enclosure 6. The plant-specific boron dilution analysis continues to show that a SFP dilution event (starting at the TS minimum value of 1800 ppm) would reasonably be recognized within 72 hours and terminated at a SFP concentration of 750 ppm, which provides ample margin to the proposed 400 ppm TS value.

In addition to satisfying the TS 4.3.1.1 criticality criteria for normal conditions, the proposed loading configurations were also analyzed for the criticality impact of abnormal and accident conditions, which are described in Enclosure 6. That analysis showed that the limiting abnormal case (i.e., fresh fuel misplacement) required a soluble boron concentration of 910 ppm, which is significantly below the TS minimum soluble boron concentration of 1800 ppm.

TS 4.3.1 is also revised to provide a complete set of loading restrictions on the storage of fuel, fuel inserts, and hardware that must satisfy the LCO of TS 3.7.17. These controls include the following:

- New Table 4.3.1-1 describes the "Fuel Categories Ranked by Reactivity" and provides some general rules for use of these categories. For instance, this table explains that any higher-numbered fuel category can be used in an array specifying a lower-numbered fuel category. It also explains special categories, such as Category 1 (for fresh unburned fuel) and Category 7 (for consolidated fuel) and the specific restrictions on those fuel categories. These fuel categories are supported by Tables 4.3.1-2 and 4.3.1-3 and align with the SFCA.
- New Table 4.3.1-2 relates specifically to fuel operated in Unit 1 and Unit 2 Cycles 1 through 4 and describes the polynomial expression and coefficients that define fuel Categories 3, 5, and 6, which are the only categories anticipated to be needed for this legacy fuel[3]. The user enters this table with fuel assembly information gained from core design records (e.g., initial enrichment), core depletion calculations (burnup), and operating records (decay time). This table also describes the use of the polynomial to calculate the minimum burnup requirement and provides rules for interpolation between values of burnup. This table is an output of the SFCA.
- New Table 4.3.1-3 relates specifically to fuel not operated in Unit 1 or Unit 2 Cycles 1 through 4 and describes the polynomial expression and coefficients that define fuel Categories 2, 3, 4, 5, and 6. This table includes a broader range of fuel categories than Table 4.3.1-2 because it represents legacy fuel as well as future fuel that may require full use of the new storage arrays. This table is entered with the same type of data used for Table 4.3.1-2 and includes similar notes. This table is an output of the SFCA.
- Figure 4.3.1-1 provides the Spent Fuel Pool Loading Restrictions. Once the fuel assemblies are categorized per the Tables in TS 4.3.1, this figure describes the loading configurations (described as "arrays") that ensure subcriticality criteria are satisfied. These proposed loading restrictions address new entities that had not been previously subject to TS: (1) consolidated rod storage canisters (CRSCs) and associated non-fissile hardware, (2) fuel rod storage canisters (FRSCs), (3) failed fuel pin baskets (FFPBs), and (4) fuel assembly inserts. In addition, one array (Array G) takes credit for the insertion of a Rod Cluster Control Assembly (RCCA) in a spent fuel assembly, whereas RCCAs had not been previously credited. This figure is an output of the SFCA.

[3] The other fuel categories (2 and 4) were created for arrays specifically associated with anticipated future core offload conditions. Note further that Category 1 is not included in the table because it represents fresh unburned fuel, and Category 7 is not included because it represents consolidated fuel that has been analyzed specifically. No further fuel consolidation is expected.

All of the loading configurations allowed by the proposed TS were specifically analyzed or evaluated to ensure that the subcriticality criteria of TS 4.3.1 were satisfied. This SFCA is provided in Enclosure 4 (non-proprietary) and Enclosure 6 (proprietary).

Human Performance Factors

This section describes the process used to preclude human performance errors associated with the placement of fresh and spent fuel in the SFP, particularly in light of the new fuel categories and loading restrictions imposed by the proposed TS and assumed in the SFCA. The main objective of this discussion is to demonstrate the continued validity of the Double Contingency Principle, which has been a regulatory basis for nuclear fuel storage criticality analyses, and states that two unlikely independent and concurrent incidents or postulated accidents are beyond the scope and need not be analyzed (Reference 6.8). A key element of this discussion is demonstrating that the existing procedures for fuel selection and placement, in conjunction with existing human performance measures will continue to provide assurance of proper fuel placement. Commitments were made to revise existing procedures and are summarized in Enclosure 7 of this LAR. The process described in this section covers: (1) fuel characterization, (2) fuel categorization, (3) the placement of fuel assemblies into the spent fuel pool storage racks, (4) the placement of other non-fuel items in the SFP storage racks, and (5) the insertion of an RCCA when required by the SFCA.

Fuel Characterization. For fuel storage arrays that rely on more than rack geometry alone to maintain subcriticality (such as those arrays in the current and proposed TS), the first critical step is to properly characterize each fuel assembly by the controlling parameters defined by TS. For TS 3.7.17 and 4.3.1, these parameters have been: initial enrichment, burnup, decay time, and gadolinium content. In the proposed TS, the parameter of gadolinium content is eliminated and the distinction by core operating cycle (pre- and post-Cycle 4) is added. Certain parameters (e.g., initial enrichment and gadolinium content) are determined from fuel procurement records. Other parameters (e.g., burnup, decay time, and operating cycle) are determined from core operating records. The value of burnup is calculated as the average assembly exposure in MWD/MTU and is currently calculated with an industry standard system called BEACON™. Other industry standard burnup calculation methods preceded BEACON, and provided results of appropriate quality.

With respect to fuel characterization, the only significant parameter change in the proposed TS is the use of Core Operating Cycle to distinguish which TS table applies to a given irradiated fuel assembly. This change causes no net increase for a human error because any risk associated with

identifying the wrong operating cycle is comparable to the current risk of mistakenly identifying a non-gadolinium assembly as a gadolinium assembly. Neither of these parameters is visually evident on a fuel assembly, but is derived from fuel or operating records and administratively tied to the particular fuel assembly serial number identification (ID). In that regard, these two different parameters (Cycle designation vs. gadolinium content) are substantially equivalent in risk from a human factors standpoint.

The output of the fuel characterization is a documented record that provides an accurate set of parameters for a given fuel assembly ID, which is then used in the ensuing processes to categorize the assembly and move it to an approved storage array.

Fuel Categorization. Once characterized, the parameters of a fuel assembly are compared to TS limits to ascertain the assembly's relative reactivity; its "category". The fuel assembly's category will then determine the storage arrays for which the assembly is qualified. The current TS do not explicitly create fuel categories because the storage configurations (arrays) are simple enough that they did not require significant variations in fuel reactivity for checkerboarding. However, to accommodate the general increase in spent fuel burnup requirements stemming from the proposed SFCA, new storage arrays using more extensive checkerboarding of a wider range of high-reactivity and low-reactivity fuel had to be constructed to continue to satisfy spent fuel storage capacity requirements.

As described in the proposed new TS Table 4.3.1-1, a fresh fuel assembly of an approved type is simply categorized as "fresh" if it meets the maximum value of initial U-235 enrichment prescribed by TS. On the other hand, a spent fuel assembly is categorized by comparing its parameters against the limits of TS Tables 4.3.1-2 (for fuel operated Cycles 1-4) or TS Tables 4.3.1-3 (for fuel not operated in Cycles 1-4). A spent fuel assembly qualifies for categories 2 – 6 based on its value of burnup exceeding the value prescribed by the respective polynomial in TS Table 4.3.1-2 or 4.3.1-3.

With respect to fuel categorization, the proposed amendment offers no significant change from current processes. Although the current TS do not describe the polynomials for determining minimum burnup requirements, in practice, the polynomials that represent the current TS curves are used rather than the TS curves themselves. In this regard, the expression of burnup limits in the proposed TS as polynomials (as opposed to curves) is not a change in the fuel categorization process.

In effect, the use of the polynomials causes no net increase for a human error because the polynomials represent more mathematical precision than

do the curves. Curves may be subject to distortions caused during reproduction and may be subject to misinterpretation based on line thickness and the visual acuity of the reader. In that regard, fuel categorization with the proposed TS is substantially equivalent in risk from a human factors standpoint.

PINGP uses the computer program ShuffleWorks™ (hereafter referred to as "ShuffleWorks") to help keep record of fuel assembly characteristics and location, and to help plan fuel movement campaigns by applying algorithms to ensure the fuel is categorized and placed in accordance with TS requirements. Once updated with the proposed TS fuel categories described by TS Table 4.3.1-1, ShuffleWorks will categorize a fuel assembly based on its new burnup requirements.

Note on Configuration Control - Positive location of a fuel assembly to its indexed SFP location. Prior to discussing the administrative controls which ensure that a fuel assembly is moved to a proper location during fuel handling campaigns, the following text discusses the controls which ensure that, as an initial datum, the proper fuel assembly (identified by serial number ID) is actually in its designated SFP location when the campaign begins. This datum is important because SFP fuel move procedures do not require a visual identification of a fuel assembly serial number ID prior to movement. Rather, fuel movement campaigns typically identify a fuel assembly only by its indexed "from" location in the SFP. Furthermore, routine spent fuel pool inventories do not systematically perform a visual identification of a fuel assembly ID to verify that a particular assembly is in a particular SFP location. Notwithstanding the above, certain procedures provide a high level of confidence that assemblies are stored safely in proper arrays as an initial datum for any subsequent move. These procedures include:

1. The core inventory procedure requires that each fuel assembly (identified by its ID) be verified in its designated core location and recorded properly in ShuffleWorks prior to reactor startup from a refueling outage. Any deviations must be reconciled and the record corrected. Thus, spent fuel assemblies discharged from the reactor core originate from a designated location that has been validated to contain the designated fuel assembly ID. Furthermore, procedures require that ShuffleWorks be updated to accurately reflect individual fuel assembly data (including burnup data and discharge date).
2. As described in more detail later, Fuel Transfer Logs methodically initiate, verify, and record the movement of any fuel assembly or fuel assembly insert within the core, between the core and SFP, and within the SFP. Each move is tracked by its indexed "from" location and its indexed "to" location, and verified by an independent observer.
3. Routine SFP inventories help confirm that fuel assemblies are in their

designated locations as recorded in ShuffleWorks. Each indexed location in the SFP is represented in ShuffleWorks with information to describe the contents of each cell, whether it be a fuel assembly, RCCA, consolidated rod storage canister (CRSC), etc., or non-fissile material such as fuel assembly structural remnants. Annual inventories required by 10 CFR 74.19(c) visually confirm that the special nuclear material listed in ShuffleWorks is actually located in the SFP. Spent fuel assemblies are verified by item count. This annual inventory is verified by a Nuclear Engineer. This annual inventory, and other routine inventories help ensure the accuracy of ShuffleWorks as a datum for fuel assembly actual locations.

Based on the logic that a spent fuel assembly was positively identified in the core by its ID, and that every move in the SFP is methodically made and verified thereafter, existing procedures provide a high level of confidence in configuration control that a particular fuel assembly accurately resides in its recorded storage location at the beginning of a fuel movement campaign. Routine inventories also support this confidence level.

Placement of Fuel Assemblies in the Spent Fuel Pool Storage Racks. As discussed previously, ShuffleWorks will continue to be the primary tool for planning and recording a fuel movement campaign in the SFP. Using ShuffleWorks, a qualified Nuclear Engineer generates a Fuel Transfer Log for the desired fuel movement campaign by selecting assemblies for movement and requesting ShuffleWorks to apply programmed rules and algorithms to determine whether a fuel assembly meets the requirements for the requested "to" location. ShuffleWorks provides a printout of the Fuel Transfer Log showing the designated "from" and "to" locations of each move. To supplement the quality of the ShuffleWorks program, the Fuel Transfer Log is prepared and verified by qualified individuals. Actual fuel movement is controlled by refueling procedures in accordance with the Fuel Transfer Log. Each move is controlled in a stepwise manner, requiring signoff for each step. Prior to the movement of fuel or fuel insert, the "from" and "to" locations are stated by the SFP Operator and the location visually verified by the Fuel Handling Supervisor or Fuel Accountability Engineer. Procedure placekeeping and three-part communications human performance tools are used to improve the integrity of this process.

For the movement of fresh fuel into the spent fuel pool and for movement of fuel within the SFP, dependent errors are prevented by the use of the Fuel Transfer Log. As described previously, fuel moves in the SFP are controlled in a stepwise manner, and procedures require that only one fuel assembly at a time shall be handled in the SFP. If a step is performed but inadvertently not signed off, attempting to perform the step again would fail because the "from" location would be empty. If a step is inadvertently

skipped, subsequent steps would be unaffected because they each contain their own "to" and "from" locations. Similarly, if an assembly is placed in the wrong location or taken from the wrong location, subsequent steps would be unaffected unless a subsequent "to" location coincided with the misplaced assembly or mistaken location. In each case, there are no dependent errors.

While the assembly identification numbers are not checked during the in-process verification or the post-campaign validations, passing the post-campaign validation with misloaded assemblies would require multiple misplacements that happened to result in the same cell vacancies as those originally planned. If a fresh fuel assembly is misplaced in the SFP while pre-staging fuel for a refueling, the error would be found either: (1) during the core offload if it was placed in a location designated for an offload assembly, or (2) during the core reload. If a reload assembly from the core is misplaced in the SFP, it would be found during the core reload, as with a misplaced fresh assembly. If a permanently discharged assembly is misplaced into a vacant cell of the SFP, the error would be discovered during a subsequent movement campaign that involved either the correct location or the actual location of that fuel assembly.

Furthermore, the prescribed operation requires verbal agreement on the indexed SFP location. Thus, a misplaced fuel assembly would have to be caused by operator error that misidentified the indexed location during bridge crane operation and by chance, landed the assembly in an empty cell. Based on the typical SFP inventory in permanent storage racks being greater than 75% full and the random chance that the misidentified location is empty, the likelihood of operators misplacing a spent fuel assembly in the wrong location is very low.

Placing new fuel in the SFP to comply with the proposed TS will not result in a change to the current process because all fuel movement continues to be controlled by the Fuel Transfer Log and associated handling procedures. In this regard, the proposed changes involve no new risk from a human factors standpoint.

Notwithstanding this conclusion, NSPM will improve procedures in this area. Current procedures require a post-campaign validation only if one or more fresh fuel assemblies are involved. To strengthen the validation of all SFP movement campaigns involving nuclear fuel, procedures will be revised to require a post-campaign validation of all affected SFP locations, whether fresh fuel is involved or not.

Offloading and Reloading Fuel into the SFP During a Refueling Outage. Core offloads and reloads are also controlled by a Fuel Transfer Log, but are different in that half of a step is performed in containment while

the other half is performed in the SFP. To preclude fuel misplacement during this operation, the offload and reload are centrally supervised by the Containment Senior Reactor Operator (hereafter referred to as "C-SRO"). The C-SRO oversees the core manipulations and the SFP moves. In support of the C-SRO, the Control Room Operator (CRO) verifies all fuel transfer moves as specified in the Fuel Transfer Log. This CRO must concur with a fuel move prior to: (a) placement of any assembly into a new core location, change fixture, or transfer car, (b) lowering the manipulator crane onto any fuel assembly, (c) placement of any fuel element into a new SFP location, and (d) lowering the spent fuel handling tool onto a fuel assembly. The CRO uses a hardcopy of the Fuel Transfer Log as well as the computer program ShuffleWorks to monitor and document each fuel movement as it is completed. Additionally, the locations for each move are verified by the appropriate field supervisor using a hardcopy of the Fuel Transfer Log. Using the C-SRO / CRO to control fuel movement serves to ensure that the "from" locations in the reactor are synchronized to the correct "to" locations in the spent fuel pool during the offload and that the "from" locations in the spent fuel pool are synchronized to the correct "to" locations in the reactor during the reload.

Because all spent fuel movement continues to be controlled by the Fuel Transfer Log and associated handling procedures, repositioning fuel in the SFP to comply with the proposed TS will not result in a change to the current process. In this regard, the proposed changes involve no new risk from a human factors standpoint.

Placement of Other Non-Assembly Fuel Items in SFP Storage Racks. Based on their small quantity and infrequent usage, the storage requirements for certain non-assembly fuel items have historically been defined in the USAR, rather than TS and procedures. These non-assembly fuel items include the: (1) consolidated rod storage canisters (CRSCs), (2) fuel rod storage canister (FRSC), and (3) failed fuel pin basket (FFPB). Based on their fissile material content, any of these items could increase the reactivity of a fuel array; however, the proposed analysis shows how these items can continue to be stored in a safe manner to meet criticality criteria without creating unacceptable human performance risks:

1. **CRSCs.** Whereas the current analysis of CRSCs imposes no restriction on their placement from a criticality standpoint (i.e., CRSCs qualify for "all-cell" storage), the additional conservatism in the proposed SFCA requires that new loading restrictions be imposed. Specifically, Array F requires a diagonally-opposed array of CRSCs, face-adjacent to empty cells (which may include non-fissile material as discussed later). Although this array involves a new level of complexity compared to the all-cell array, there is no practical increase in the risk of a human performance error because: (a) the CRSCs will not have to be moved

to comply with the proposed new restrictions (i.e., they are currently checkerboarded), and (b) there is no impetus to move CRSCs in the future. CRSCs have been dedicated to one particular storage rack in the SFP and are not moved because of limitations associated with seismic qualification of the racks and because their weight imposes special lifting requirements.

2. **FRSC.** As described in USAR Section 10.2.1, special storage requirements are assigned to the FRSC based on the criticality analysis; relating the FRSC's reactivity to that of an assembly of equal enrichment. The USAR states that the FRSC can be stored as a fresh fuel assembly and does not state that it would qualify for the all-cell configuration. However, the proposed SFCA specifically analyzed the FRSC and concluded that it could be stored in any storage cell where a fuel assembly would be allowed. Thus, the new analysis supports less restrictive storage requirements for the FRSC, which reduces the risk of a human performance error that would adversely affect reactivity.
3. **FFPB.** Both the current analysis and the proposed analysis indicate that these items may continue to be substituted for a fuel assembly in any approved storage array. In that respect, the proposed analysis does not create any new human performance challenge or restriction.

Noting the small quantity of these non-assembly fuel items in storage, and that the proposed amendment provides no impetus to move these items within the SFP, no adverse human factors considerations are created by the proposed amendments. In this regard, the proposed changes involve no new risk from a human factors standpoint.

Nevertheless, to maintain consistency for placing criticality controls for fissile material in the TS (per 10 CFR 50.36), the proposed TS include restrictions on the placement of these items in the SFP. These TS will help elevate the awareness of these requirements, improve compliance, and thereby reduce the risk of human error to misplace one of these items.

Placement of Other Material in SFP Storage Racks. Based on their small quantity, no impetus for movement, and no stated restrictions in the current criticality analysis, there are no current limitations on the placement of other non-fuel materials to preserve subcriticality. These materials include neutron source assemblies, incore detectors, fuel assembly hardware, other irradiated metal debris, and dummy fuel assemblies. Any of these materials could increase the reactivity of a fuel array, particularly if the material serves as a more efficient fission-neutron reflector than that assumed in the analysis. Restrictions and human performance factors associated with these items are discussed below:

1. **Neutron Source Assemblies**. These neutron sources are described in the PINGP Final Safety Analysis Report (FSAR) Section 3.2. As

described in the USAR Section 3.5.2.2, these fuel assembly inserts are no longer used in the reactor core. Although the current USAR does not specifically evaluate the criticality effect of a neutron source in the SFP and current TS do not restrict its placement in the SFP, the proposed SFCA and TS changes do address it. The SFCA evaluated the reactivity effect of components inserted into fuel assemblies and concluded in Section 5.4.1 that any fuel assembly insert approved for use in the reactor core (which includes a neutron source) would be acceptable to store in a fuel assembly in the SFP storage racks. Therefore, the proposed analysis supports no restriction on the placement of neutron sources in the SFP, and the proposed amendment creates no change to the risk that a misplacement of these sources would challenge the criticality criteria.

2. **Hardware in Consolidated Fuel Storage (New Array F)**. During the Spent Fuel Consolidation Project described in USAR 10.2.1.5, CRSCs and "assembly cage components" were placed into a particular 7x8 storage rack in a checkerboard pattern. The volume of the assembly hardware was reduced during the project. This checkerboard pattern was not required by the prevailing criticality analysis at the time, but it helped to distribute the weight of the CRSCs equally in the storage rack. In the new proposed SFCA, Array F has been conservatively analyzed with this hardware in the storage array to fill the otherwise "empty cells". Based on the conservative nature of this calculation bounding the foreseeable maximum density of hardware in a storage cell, and noting that the human factors associated with placement of CRSCs is discussed above, it is shown that the allowance for hardware in new Array F does not involve a new risk from a human factors standpoint.
3. **Hardware in New Arrays B - E**. Current TS do not offer a storage array that specifically requires an empty cell; therefore, placement of non-fissile hardware into an empty cell of a storage array has historically had no potential effect on SFP criticality margins. However, the new SFCA and TS offer four new arrays (Arrays B through E) that specifically require an empty cell that must not contain any hardware. In this regard, the proposed amendment offers a new human performance challenge to plant operators: placement of otherwise inconsequential hardware in a vacant cell may now have an adverse reactivity effect. To address this challenge, proposed TS (Note 5 of Figure 4.3.1-1) specifically states that the empty cells of Arrays B – E must be empty. Also, NSPM will revise plant procedures to ensure hardware moves (including dummy assemblies) are supervised, monitored, and recorded in a manner comparable to that used for fuel assemblies (e.g., using Fuel Transfer Log) and include a visual verification at the conclusion of the campaign to ensure the safe location of the relocated hardware.

Historically, there is little cause to move most of these hardware items within the SFP (e.g., the assembly cage hardware in the consolidated fuel

rack), primarily because any movement creates the risk of generating foreign material in fresh fuel assembly. The notable exception to this position is the dummy test-weight fuel assembly which is used routinely to test the spent fuel pool bridge crane and for training purposes. To address the use of the dummy, its movement in the SFP is included in the commitment to revise handling procedures for non-assembly material.

In summary, two factors ensure the net effect of the proposed amendments will not create an unaddressed human factor consideration that would adversely affect the conditions assumed in the spent fuel criticality analysis: (1) the proposed Technical Specifications specifically constrain the placement of the aforementioned non-fuel hardware, and (2) plant procedures will be revised to control the placement of such hardware particularly to ensure it is not placed in a cell designated to be empty.

Insertion of RCCA When Required by the SFCA. Current TS and SFCA do not credit RCCAs for any storage array. Nonetheless, plant procedures have maintained positive control of RCCA moves within the SFP. RCCA moves are identified, verified, supervised, and recorded in a manner that is comparable to the previously-described fuel assembly movement procedure (i.e., Fuel Transfer Logs).

The proposed TS and SFCA will take credit for RCCAs in the center assembly of Array G. Thus, any inadvertent move of an RCCA from its required location would affect the criticality margin of the analysis and must be prevented. Accordingly, the existing procedural controls on RCCA movement in the SFP will continue to ensure proper placement with the same assurance that is provided for proper fuel assembly placement. Therefore, implementation of the proposed new storage array that credits an RCCA inserted into a center assembly does not involve a new risk from a human factors standpoint.

In summary, existing procedures will ensure that the net effect of implementing a configuration that credits RCCA placement does not create a new human factors risk. Multiple verification of the "from" and "to" locations used during actual RCCA movement along with independent verification of an RCCA's resultant storage location will maintain the same level of confidence in human performance as that provided for fuel assembly movement.

Thus, the level of safety prescribed in the regulation (10 CFR 50.68(b)) and related regulatory guidance continues to be satisfied with the proposed amendment with consideration of the methodology, inputs, and uncertainties provided in the guidance of the Draft ISG.

3.3.2 Justification for Spent Fuel Criticality Analysis Methods

The analysis methods are described in the SFCA (Enclosures 4 and 6) and approval is requested herein. These methods are justified because:

- key analytical codes used in the analysis are topically approved (as described in Section 2.3 of the SFCA),
- inputs are conservatively selected and applied (as described in Section 3 of the SFCA),
- biases and uncertainties are conservatively applied (as described in Section 4.1.2 of the SFCA), and
- results satisfy the regulatory criteria of 10 CFR 50.68(b) (as described in Section 5 of the SFCA).

As discussed in the Regulatory Evaluation Section of this Enclosure (Section 4.2), the Draft ISG provides guidance for an acceptable SFCA. Each of the major topics of the Draft ISG is reviewed below with some of the salient points that help explain how the proposed SFCA meets that guidance topic:

1. Fuel Assembly Selection. As the guidance suggests, the SFCA assessed the current fuel design and all the legacy fuel designs used at PINGP to establish the limiting fuel design over a full range of burnup values. Refer to the SFCA Section 3.1.
2. Depletion Analysis. The depletion analysis used in the proposed SFCA applies the conservative approach of the Draft ISG to generate isotopic number densities and is described predominantly in SFCA Section 3.3. Conservative reactor operating parameters were selected to correct previous nonconservatisms in the PINGP analysis of record, and to align with the Draft ISG. For example, it was assumed that each assembly operates at an assembly average relative power of 1.536 as described in Section 3.3.2.2 of the SFCA. Also, full power operation with control rods inserted (also known as "rodded operation"), was analyzed with a conservative assumption of 1 gigawatt-day per metric ton uranium (GWd/MTU) period of rodded operation at full power to provide a basis to bound historical and future operation (Reference SFCA Section 4.3). To supplement this analysis, NSPM also commits to assess any past or future rodded operation that exceeds 1 GWd/MTU as described in Enclosure 7, Commitments 2 and 3. Lastly, the SFCA calculates and applies a depletion uncertainty as suggested in the Draft ISG. Refer to SFCA Section 4.1.2.1.3.
3. Criticality Analysis. A salient topic of the Draft ISG is the assessment of all "normal conditions" in the SFCA which go beyond a fuel assembly in its stored location. To meet this expectation, the proposed SFCA specifically evaluated the following types of "normal conditions": (1) Placement of components such as control rods and thimble plugs into or

near to fuel assemblies, (2) Evolutions involving a fuel assembly outside the storage rack including cleaning and sipping, (3) Placement into the storage racks of components that are not intact fuel assemblies (e.g., consolidated fuel, damaged fuel, movable in-core detectors), (4) Temporary placement around the storage rack periphery of components such as sipping cans, and (5) Miscellaneous conditions such as damaged storage cells and debris under the storage racks. Refer to SFCA Sections 3.6 and 4.4.

4. Criticality Code Validation. Appendix A ("Validation of SCALE 5.1") of the SFCA provides the analysis of data trends and the area of applicability suggested by the Draft ISG. As suggested in the Draft ISG, this analysis included the Haut Taux de Combustion (HTC) experiments using mixed oxide fuel.
5. Miscellaneous. As suggested in the Draft ISG, assumptions used in the SFCA are explicitly enumerated and justified, predominantly in Section 3 of the SFCA.

In summary, the SFCA was prepared to meet the guidance of the Draft ISG and was accepted as a quality document under NSPM's 10 CFR 50 Appendix B Quality Assurance Program.

3.3.3 Justification for Technical Specification Changes - Elimination of Part 72 Cask Soluble Boron Requirements

With respect to the soluble boron concentration requirements associated with cask loading restrictions (TS 4.3.1.3), note that this TS was added in 1992 by PINGP License Amendments 99/92. Subsequently, 10 CFR 50.68 was revised to clarify that such a Part 50 TS was not required because the fuel criticality in the cask was covered by Part 72 regulations. 10 CFR 50.68(c) currently states: "While a spent fuel transportation package approved under Part 71 of this chapter or spent fuel storage cask approved under Part 72 of this chapter is in the spent fuel pool, the requirements in § 50.68(b) do not apply to the fuel located within that package or cask; and the requirements in Part 71 or 72 of this chapter, as applicable, and the requirements of the Certificate of Compliance for that package or cask, apply to the fuel within that package or cask." In compliance with Part 72, the soluble boron requirements for cask subcriticality are prescribed in the Prairie Island site-specific ISFSI license SNM-2506 (Reference 6.3). The associated ISFSI Technical Specifications (3.3.1, Dissolved Boron Concentration) specifically establish the soluble boron requirements during loading and unloading operations. Thus, having established the cask-loading soluble boron requirements in the appropriate Part 72 license, it is justified to remove those redundant requirements from the Part 50 license as allowed by 10 CFR 50.68(c).

3.3.4 Justification for Licensing Basis Change – Adoption of 10 CFR 50.68(b)

Another proposed current licensing basis change relates to the regulatory basis for the SFCA. Currently, NSPM complies with 50.68(a) through the option to comply with 10 CFR 70.24, Criticality Accident Requirements. Accordingly, NSPM performs criticality monitoring and criticality accident evacuation drills required by 70.24(a). In the absence of criticality criteria in 10 CFR 70.24, NSPM had previously selected the criticality criteria that are now expressed in 50.68(b) and incorporated those criteria into Technical Specifications. To clarify this regulatory basis and eliminate the regulatory obligations of 10 CFR 70.24, the proposed amendment will obligate NPSM to 10 CFR 50.68(b) and invoke all requirements therein, including revision to the Updated Final Safety Analysis Report (UFSAR) per 50.68(b)(8). In effect, this licensing basis change imparts no change to the PINGP SFCA acceptance criteria because those criteria have been, and will continue to be, those described by 10 CFR 50.68(b).

3.4 Conclusion

The criticality analysis addresses each of the storage configurations described by the proposed TS and confirms that the effective neutron multiplication factor (k_{eff}) of all permissible fuel storage arrangements is less than 1.0 when the pool is assumed to be flooded with unborated water. Also, for the limiting configuration, the analysis computes the value of SFP boron concentration (359 ppm) that would ensure all normal fuel configurations experience a k_{eff} less than or equal to 0.95. Review of plant-specific boron dilution analysis shows that a SFP dilution event (starting at the TS minimum value for SFP boron concentration) would be recognized and terminated before the value of 359 ppm is approached. Finally, the analysis also demonstrates that the k_{eff} is less than 0.95 for all postulated abnormal and accident conditions, taking credit for soluble boron concentration (as high as 910 ppm).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The US Atomic Energy Commission (AEC) issued their Safety Evaluation (SE) of the PINGP on September 28, 1972. The SE, Section 3.1, "Conformance with AEC General Design Criteria," described the conclusions the AEC reached associated with the General Design Criteria in effect at the time. The AEC stated:

The Prairie Island plant was designed and constructed to meet the intent of the AEC's General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 50% complete and the Final Safety Analysis Report (Amendment No. 7) had been filed with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the

criteria in July 1971. We did not require the applicant to reanalyze the plant or resubmit the FSAR. However, our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria.

Based on the above, the applicable PINGP GDC states: Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

In the course of licensing history of PINGP, NSPM has maintained a regulatory commitment to 10 CFR 70.24, "Criticality Accident Requirements", which requires operation of criticality monitors (70.24(a)(2)) and performance of criticality accident evacuation drills (per 70.24(a)(3)).

10 CFR 50.68(c) states: "While a spent fuel transportation package approved under Part 71 of this chapter or spent fuel storage cask approved under Part 72 of this chapter is in the spent fuel pool, the requirements in § 50.68(b) do not apply to the fuel located within that package or cask; and the requirements in Part 71 or 72 of this chapter, as applicable, and the requirements of the Certificate of Compliance for that package or cask, apply to the fuel within that package or cask."

4.2 Precedent

In August 2010, NRC issued Draft Interim Staff Guidance DSS-ISG-2010-01, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools (Reference 6.1) to rebaseline NRC's expectations for spent fuel criticality analysis. That guidance was intended to reiterate existing guidance, clarify ambiguity in existing guidance, and indentify lessons learned based on recent submittals. Further, Section 5 (entitled Miscellaneous) of the Draft ISG included specific caution for applying precedent. The expectations of the Draft ISG were further reinforced in subsequent NRC Information Notice 2011-03 (Reference 6.4).

Based on the new NRC baseline guidance and the caution for use of precedence on this topic, little precedent is applicable to this LAR. The only related submittals to have been submitted on this topic after issuance of the Draft ISG were the Turkey Point LAR No. 207 (Reference 6.5) and the Palisades LAR (Reference 6.6); both of which are currently under NRC Staff review. Both of the above submittals were actually supplements to applications that pre-dated the Draft ISG. However, the Turkey Point supplement acknowledged the Draft ISG and its supporting analysis (performed by Westinghouse) addressed elements of the Draft ISG.

NSPM has employed the same Westinghouse organization to perform the PINGP SFCA that performed the Turkey Point analysis. NSPM also performed an audit to

understand any significant differences between the Turkey Point and PINGP analyses, and has incorporated any necessary results. Noting that the PINGP analysis compares well to the Turkey Point analysis methods and no Requests for Additional Information (RAIs) have been formally issued for the proposed Turkey Point amendment (as of July 13, 2011), no particular precedent is established from that submittal.

4.3 Significant Hazards Consideration

Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, proposes to amend the facility operating licenses of Prairie Island Nuclear Generating Plants (PINGP) Units 1 and 2. The purpose of this amendment is to modify the PINGP Technical Specifications (TS) to provide new loading restrictions which maintain subcriticality in the spent fuel pool and to remove spent fuel pool soluble boron concentration requirements related to dry cask loading.

NSPM 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 change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   The proposed amendments do not change or modify the fuel, fuel handling processes, fuel storage racks, number of fuel assemblies that may be stored in the spent fuel pool (SFP), decay heat generation rate, or the SFP cooling and cleanup system. The proposed amendment was evaluated for impact on the following previously-evaluated events and accidents: (1) fuel handling accident (FHA), (2) fuel assembly misloading, (3) seismically-induced movement of spent fuel storage racks, (4) loss of spent fuel pool cooling, and (5) spent fuel boron dilution.

   Although implementation of the proposed amendment will require handling of fuel assemblies to achieve the new configurations, the probability of a FHA is not increased because the implementation of the proposed amendment will employ the same equipment and procedures to handle fuel assemblies that are currently used. Therefore, the proposed amendments do not increase the probability for occurrence of a FHA. In that the proposed amendment does not involve changes to the radiological source term of any fuel assembly, the amendment would not increase the radiological consequences of a FHA. With regard to the potential criticality consequences of a dropped assembly coming to rest adjacent to a storage rack or on top of a storage rack, the results are bounded by the fuel assembly misloading event which is analyzed to provide

sufficient margin to criticality. The fuel configuration caused by a dropped assembly resting on top of loaded storage racks is inherently bounded by the assembly misloaded in the storage rack because the misloaded assembly is in closer proximity to other assemblies along its entire fuel length.

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 worst-case fuel misloading event.

Operation in accordance with the proposed amendment will not change the probability of occurrence of a seismic event, which is considered an Act of God. Also, the consequences of a seismic event are not changed because the proposed amendment involves no change to the types of material stored in SFP storage racks or their mass. In this manner, the forcing functions for seismic excitation and the resulting forces are not changed. Also, particular to criticality, the supporting criticality analysis takes no credit for gaps between rack modules so any seismically-induced movement of racks into a closer proximity would not result in an unanalyzed condition with consequences worse than those analyzed. In summary, the proposed amendment will not increase the probability or consequence of a seismic event.

Operation in accordance with the proposed amendment will not change the probability of a loss of spent fuel pool cooling because the change in fuel loading configurations has no bearing on the systems, structures, and components involved in initiating such an event. The proposed amendment does not change the heat load imposed by spent fuel assemblies nor does it change the flow paths in the spent fuel pool. Finally, a new criticality analysis of the limiting fuel loading configuration confirmed that the condition would remain subcritical at the resulting temperature value. Therefore, the accident consequences are not increased for the proposed amendment.

Operation in accordance with the proposed amendment will not change the probability of a boron dilution event because the change in fuel loading configurations has no bearing on the systems, structures, and components involved in initiating or sustaining the intrusion of unborated water to the spent fuel pool. The consequences of a boron dilution event are unchanged because the proposed amendment has no bearing on the systems that operators would use to identify and terminate a dilution event. Also, implementation of the proposed amendment will not affect any of the other key parameters of the boron dilution analysis which includes SFP water inventory, volume of SFP contents, initial boron concentration requirement, and the sources of dilution

water. Finally, a new criticality analysis of the limiting fuel loading configuration confirmed that the dilution event would be terminated at a soluble boron concentration value that ensured a subcritical condition.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No

The proposed amendments involve new SFP loading configurations for current and legacy fuel designs of the nuclear plant. The proposed amendments do not change or modify the fuel, fuel handling processes, fuel storage racks, number of fuel assemblies that may be stored in the pool, decay heat generation rate, or the spent fuel pool cooling and cleanup system. As such, the proposed changes introduce no new material interactions, man-machine interfaces, or processes that could create the potential for an accident of a new or different type. This determination is based on the review of the two significant SFP loading changes proposed by the amendment: (1) new storage arrays, and (2) use of Rod Cluster Control Assemblies (RCCAs) in one new proposed array.

Operation with the proposed fuel storage arrays will not create a new or different kind of accident because fuel movement will continue to be controlled by approved fuel handling procedures. These procedures continue to require identification of the initial and target locations for each fuel assembly that is moved. There are no changes in the criteria or design requirements pertaining to fuel storage safety, including subcriticality requirements, and analyses demonstrate that the proposed storage arrays meet these requirements and criteria with adequate margins. Thus, the proposed storage arrays cannot cause a new or different kind of accident.

Implementation of the proposed new storage array that credits an RCCA inserted into a center assembly does not create the potential for a new or different type of accident because the operation is controlled with procedural controls comparable to those used for fuel assembly placement in the SFP and because the inadvertent RCCA removal was explicitly evaluated in the revised criticality analysis. RCCAs are installed in spent fuel assemblies in accordance with approved procedures, and movement is controlled in accordance with approved fuel transfer logs that identify and then independently verify their placement. The inadvertent removal of an RCCA from an array has been evaluated with acceptable results. The effects are bounded by the fuel assembly misloading event. Thus, the use of RCCAs in the proposed array does not create the possibility of a new or different kind of accident.

3. **Do the proposed changes 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 10 CFR 50.68 (b)(4) is unchanged. The new criticality analysis confirms that operation in accordance with the proposed amendment continues to meet the required subcriticality margin. Also, revised loading restrictions in the proposed TS have actually reduced the soluble boron requirements for the limiting normal configuration, thereby increasing the margin for the postulated boron dilution event. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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.

5.0 ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a facility requires no environmental assessment if the operation of the facility in accordance with the proposed amendment does not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (3) result in a significant increase in individual or cumulative occupational radiation exposure. NSPM has reviewed this LAR and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment. The basis for this determination follows.

1. As demonstrated in the 10 CFR 50.92 evaluation, the proposed amendment

does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. Implementation of the proposed project only involves one physical activity: establishing new location criteria for nuclear fuel, RCCAs, and certain other non-fuel material in the Spent Fuel Pool. To implement the new arrays of fuel, the fuel assemblies will be handled using established procedures during the implementation campaign. A small amount of solid low-level radioactive waste is expected to be generated during the campaign. However, this quantity of waste is expected to be less than the amount generated by a typical refueling outage, based on the estimated quantity of moves for the campaign. Otherwise, performing the fuel movement campaign is not expected to generate any gaseous or liquid effluent that would not otherwise be generated in the course of routine spent fuel pool operations over its lifetime. Also, some waste filters may be generated as a result of vacuuming the moved fuel assemblies; however, that waste and those operations would have to transpire sometime in the future as a result of normal fuel handing or cask loading operations.
3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. Implementation of the proposed amendment will involve a campaign of fuel movements with personnel in the SFP area. Aside from the small amount of individual and cumulative occupational radiation exposure resulting from that campaign, the proposed amendments will not result in any unusual spent fuel pool operations that would result in a permanent effect to increase occupational exposure. The proposed fuel storage configurations do not fundamentally change the inventory or radiological source term of the spent fuel. In addition, based on NSPM's experience with routine fuel movement campaigns during refueling outages, the cumulative exposure from the proposed activities is expected to be minimal.

6.0 REFERENCES

6.1 Draft Interim Staff Guidance DSS-ISG-2010-01, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools, dated August 10, 2010 (ADAMS Accession No. ML102220518)

6.2 Prairie Island Units 1 and 2 Operating License Amendment Nos. 99 / 92 and NRC SER dated July 9, 1992

6.3 PINGP ISFSI License No. SNM-2506 through Amendment 7 (ADAMS Accession No. ML110740182)

6.4 NRC Information Notice 2011-03, Nonconservative Criticality Safety Analyses for Fuel Storage, dated February 16, 2011 (ADAMS Accession No. ML103090055)

6.5 FPL letter to NRC, Turkey Point Units 3 and 4, License Amendment Request No. 207 Supplement 1 to Fuel Storage Criticality Analysis, dated February 22, 2011 (ADAMS Accession No. ML110560335)

6.6 Entergy letter to NRC, License Amendment Request for Spent Fuel Pool Region I Criticality, Palisades Nuclear Plant, dated January 31, 2011 (ADAMS Accession No. ML110380083)

6.7 NRC letter to FirstEnergy Nuclear Operating Company, Beaver Valley Power Station, Unit No. 2 – Issuance of Amendment Regarding the Spent Fuel Pool Rerack, dated April 29, 2011 (ADAMS Accession No. ML110890844)

6.8 NRC Memorandum, Kopp to Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", dated August 19, 1998. The "Kopp Letter".

Enclosure 2
Marked-Up Technical Specification Pages

15 pages follow

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Pool Storage

, fuel insert, or hardware

LCO 3.7.17 ~~The combination of initial enrichment, burnup and decay time of each~~ fuel assembly stored in the spent fuel pool shall ~~be within the Unrestricted Region of Figure 3.7.17-1 or in accordance with~~ Specification 4.3.1.1.

Each

satisfy the loading restrictions of

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 location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the ~~initial enrichment, burnup and decay time of the~~ fuel assembly ~~is in accordance with Figure 3.7.17-1 or~~ , fuel insert, or other hardware placed in the spent fuel storage racks is stored in accordance with Specification 4.3.1.1.	Prior to storing or moving the fuel assembly , fuel insert, or other hardware
SR 3.7.17.2	Verify spent fuel pool inventory.	Within 7 days after completion of a spent fuel pool fuel handling campaign

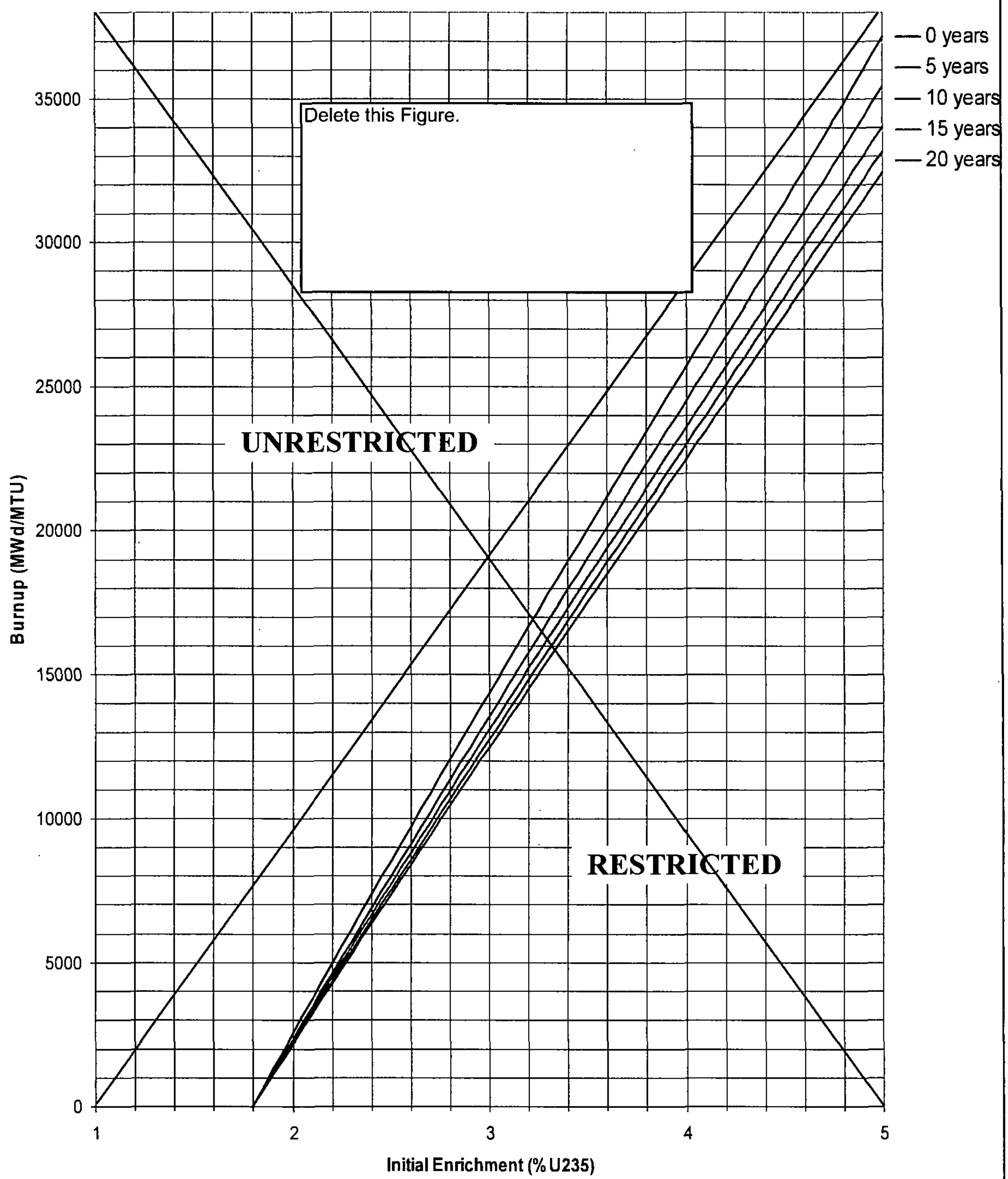


Figure 3.7.17-1
Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

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

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

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described USAR Section 10.2;

400

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to ~~730~~ ppm, which includes an allowance for uncertainties as described in USAR Section 10.2;

d. A nominal 9.5 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies ~~with a combination of discharge burnup, initial enrichment and decay time in the "unrestricted range" of Figure 3.7.17-1 may be allowed unrestricted storage in the fuel storage racks; and~~

~~f.~~ ~~New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "restricted range" of Figure 3.7.17-1 will be stored in compliance with Figures 4.3.1-1 through 4.3.1-4.~~

, fuel inserts, and hardware loaded in accordance with Figure 4.3.1-1.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

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

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

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in USAR Section 10.2;

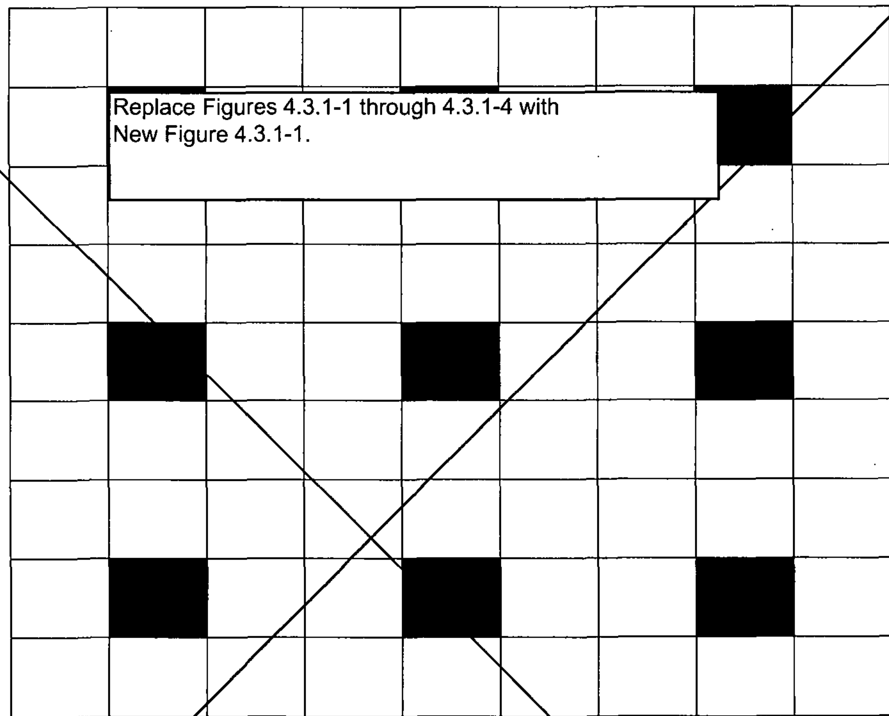
c. $k_{eff} \leq 0.98$ if accidentally filled with a low density moderator which resulted in optimum low density moderation conditions; and

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

~~4.3.1.3~~ ~~Fuel will not be inserted into a TN-40 spent fuel cask in the pool unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that~~ $\cancel{k_{eff}}$ ~~for the spent fuel cask, including statistical uncertainties, will be~~ $\cancel{\leq 0.95}$ ~~for all postulated arrangements of fuel within the cask. The criticality analyses for the TN-40 spent fuel storage cask were based on fresh fuel enriched to 3.85 weight percent U-235.~~

4.3.2 Drainage

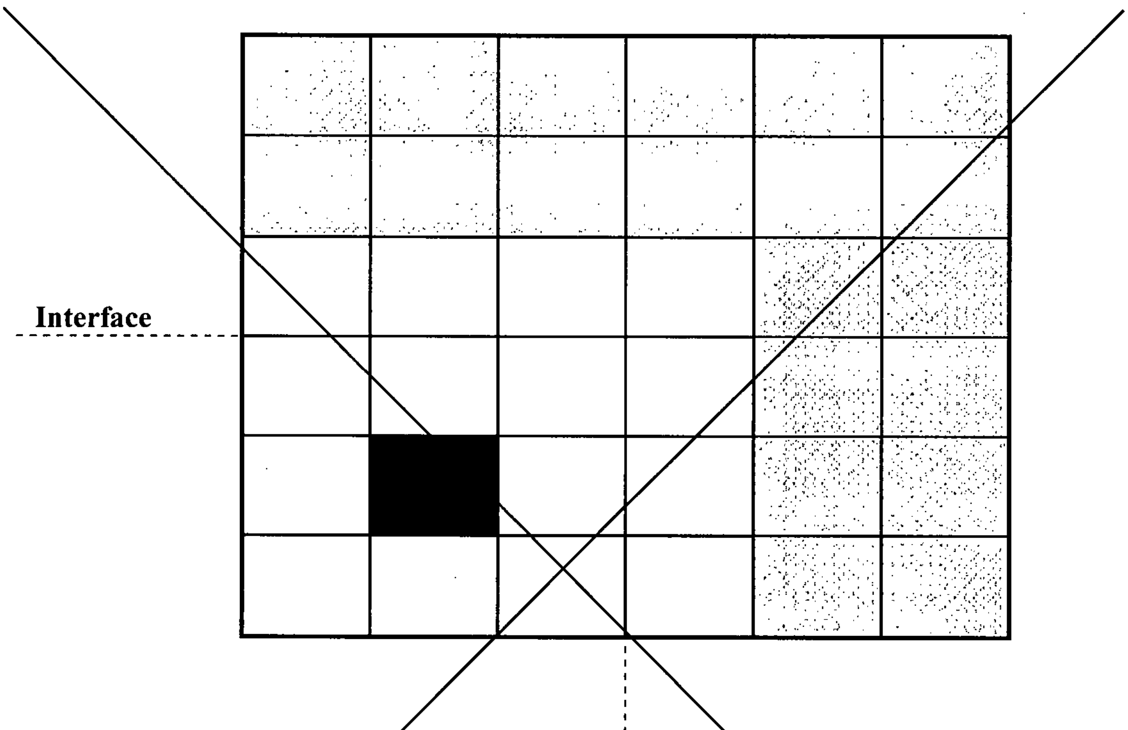
The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 727′ 4″ (Mean Sea Level).



Fresh Fuel: Must be less than or equal to Nominal 4.95 w/o ^{235}U
No restrictions on burnup
Assemblies with GAD shall have a minimum of 4 fuel rods with a minimum concentration of 4.0 w/o Gd_2O_3.

Burned Fuel: Must satisfy minimum burnup requirements of Figures 4.3.1-3 or 4.3.1-4 depending on presence of GAD rods in fresh fuel

Figure 4.3.1-1
Spent Fuel Pool Burned/Fresh Checkerboard Cell Layout



Fresh Fuel: Must be less than or equal to nominal 4.95 w/o ^{235}U
No restrictions on burnup

Burned Fuel: 3x3 Checkerboard Region
Must satisfy minimum burnup requirements of Figure 4.3.1-3 or 4.3.1-4

Burned Fuel: All Cell Unrestricted Region
Must satisfy minimum burnup requirements of Figure 3.7.17-1

Figure 4.3.1-2
Spent Fuel Pool Checkerboard Interface Requirements

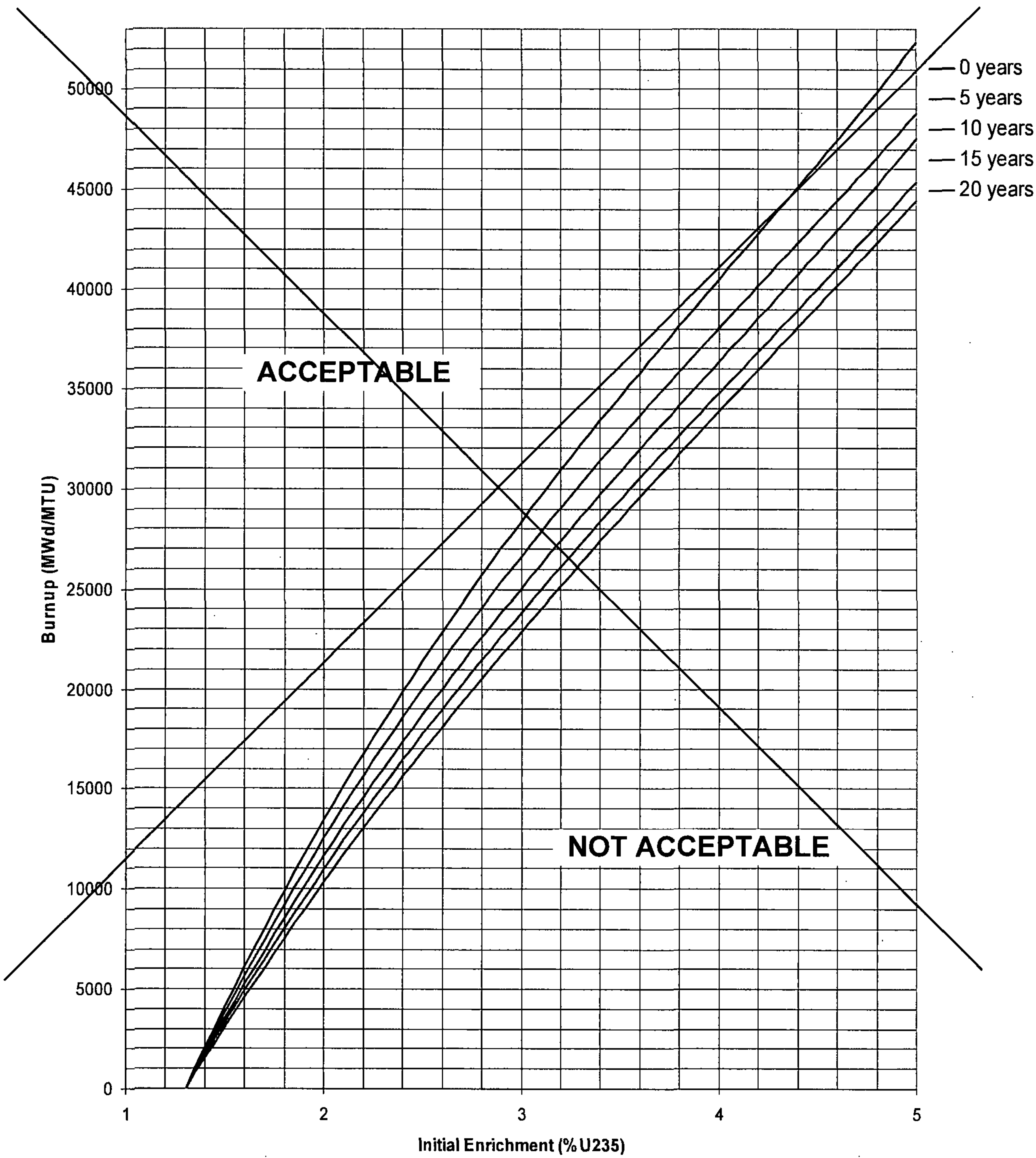


Figure 4.3.1-3
Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements
- No GAD

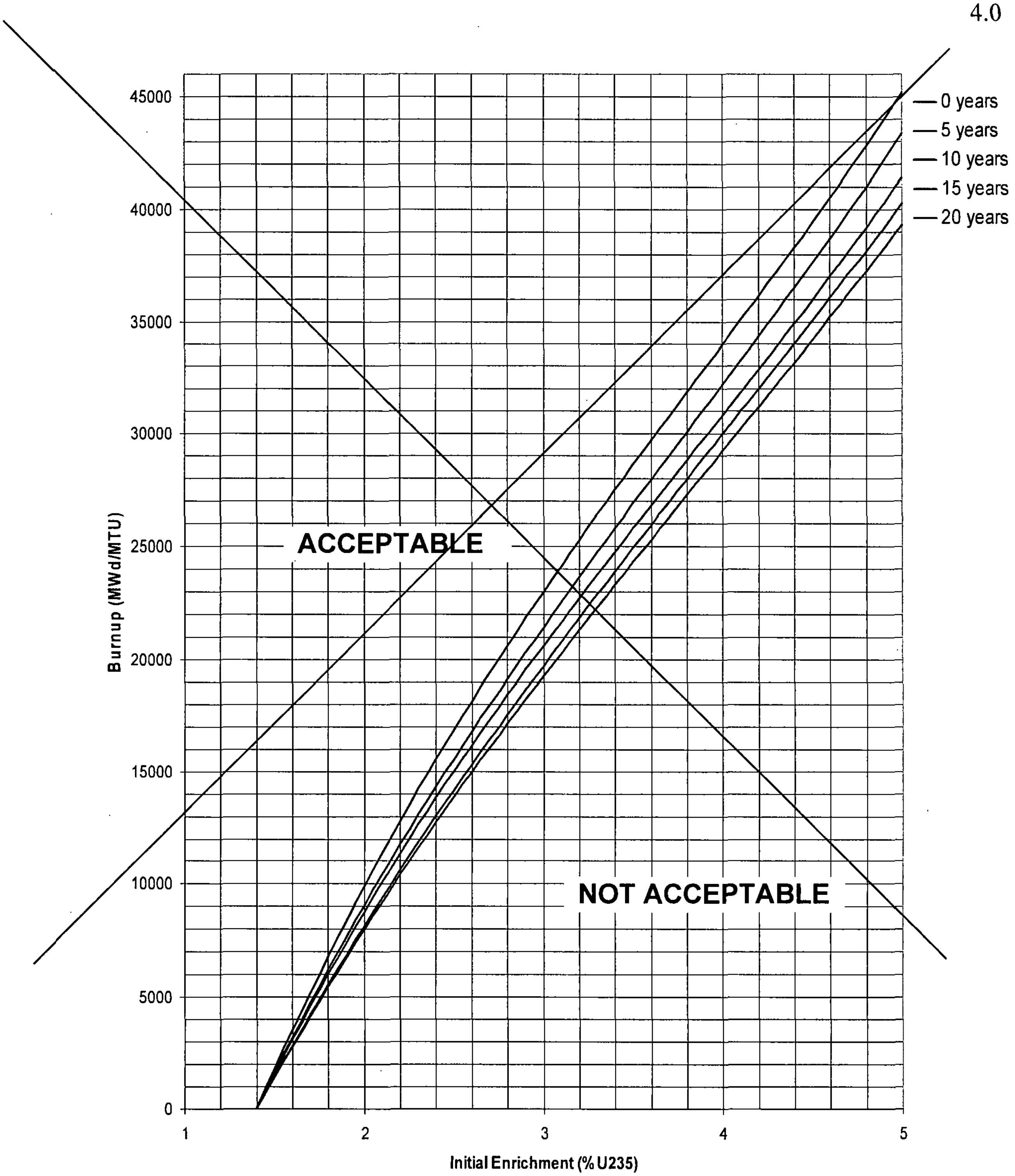


Figure 4.3.1-4
Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - Fuel with GAD

Table 4.3.1-1

Fuel Categories Ranked by Reactivity

See notes below for use of Table 4.3.1-1

Fuel Category	Relative Reactivity
1	High
2	↓
3	
4	
5	
6	Low
7	Consolidated Fuel

Notes:

1. Fuel category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category 2 is less reactive than Category 1, etc. The more reactive fuel categories require additional measures to be placed on fuel placement in the SFP racks, e.g., more use of water-filled cells or Rod Control Cluster Assemblies (RCCAs).
2. Any higher-numbered fuel category (except Category 7) may be used in an array specifying a lower-numbered fuel category.
3. Category 1 is fuel up to 5.0 weight percent U-235 enrichment and does not credit burnup.
4. Category 7 is consolidated fuel stored in Consolidated Rod Storage Canisters.
5. Categories 2 through 6 are determined from Tables 4.3.1-2 and 4.3.1-3.

Table 4.3.1-2

For Fuel Operated in Units 1 & 2 Cycles 1 - 4
Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Enrichment (En)

Fuel Category	Decay Time	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.000	-0.722	14.272	-31.167
	20	0.000	-1.944	20.494	-39.085
5	0	0.673	-8.242	44.607	-56.428
	20	1.784	-16.297	60.035	-64.713
6	0	1.097	-10.246	47.457	-56.456
	20	1.820	-15.656	56.856	-60.351

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment". The specific minimum burnup required for each fuel assembly is calculated from the following equation for each increment of decay time:

$$Bu = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment (En) is the nominal U-235 enrichment. Any enrichment between 1.7 and 3.4 weight percent U-235 may be used. If the computed Bu value is negative, zero shall be used.

3. Decay Time is in years. An assembly with a cooling time greater than 20 years must use 20 years. No extrapolation is permitted.

4. If Decay Time value falls between increments of the table, the lower Decay Time value shall be used or a linear interpolation may be performed as follows: Compute the Bu value using the coefficients associated with the Decay Time values that bracket the actual Decay Time. Interpolate between Bu values based on the increment of Decay Time between the actual Decay Time value and the computed Bu results.

5. This table applies to fuel assemblies that were operated in the core for any period of time during Unit 1 or Unit 2 Cycles 1 through 4.

Table 4.3.1-3

For Fuel Not Operated In Units 1 & 2 Cycles 1 - 4
Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Enrichment (En)

Fuel Category	Decay Time	Coefficients A_1	A_2	A_3	A_4
2	0	-0.669	9.018	-32.080	33.507
3	0	-0.120	1.300	5.006	-18.765
	5	-0.167	1.766	3.085	-16.141
	10	-0.218	2.249	1.405	-14.163
	15	-0.281	2.949	-1.267	-10.873
	20	-0.401	4.237	-5.881	-5.513
4	0	1.355	-14.866	62.715	-72.624
5	0	0.569	-6.563	37.088	-47.854
	5	0.302	-3.795	27.410	-37.964
	10	0.151	-2.248	21.874	-32.204
	15	-0.198	1.133	11.031	-21.713
	20	-0.427	3.424	3.614	-14.522
6	0	0.567	-6.205	35.936	-45.944
	5	0.923	-9.720	45.538	-53.858
	10	0.728	-7.992	40.264	-48.929
	15	0.343	-4.016	27.236	-36.380
	20	0.283	-3.391	24.925	-33.963

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment". The specific minimum burnup required for each fuel assembly is calculated from the following equation for each increment of decay time:

$$Bu = A_1{*}En^3 + A_2{*}En^2 + A_3{*}En + A_4$$

2. Initial enrichment (En) is the nominal U-235 enrichment. Any enrichment between 1.7 and 5.0 weight percent U-235 may be used. If the computed Bu value is negative, zero shall be used.
3. Decay Time is in years. An assembly with a cooling time greater than 20 years must use 20 years. No extrapolation is permitted.
4. If Decay Time value falls between increments of the table, the lower Decay Time value shall be used or a linear interpolation may be performed as follows: Compute the Bu value using the coefficients associated with the Decay Time values that bracket the actual Decay Time. Interpolate between Bu values based on the increment of Decay Time between the actual Decay Time value and the computed Bu results.
5. This table applies to fuel assemblies that were not operated in the Unit 1 or Unit 2 core during operating Cycles 1 through 4.

Figure 4.3.1-1

Spent Fuel Pool Loading Restrictions

Any fresh fuel, irradiated fuel, or non-fuel material shall meet the following restrictions prior to placement in the Spent Fuel Pool storage racks when any fuel is in the spent fuel pool:

A. Any array of storage cells containing fuel shall comply with the storage patterns in Figure 4.3.1-1 and the requirements of Tables 4.3.1-1, 4.3.1-2, and 4.3.1-3 as applicable. The category number of fuel assemblies selected for a 2x2 or 3x3 array (category determined using Table 4.3.1-2 or 4.3.1-3) shall be equal to or greater than the category number shown in the respective figure.

B. Any storage array location designated for a fuel assembly may be replaced with a failed fuel basket (fuel rod storage canister or failed fuel pin basket), incore detectors, or other non-fissile hardware.

C. Fuel assembly inserts designed for use in the reactor core may be inserted in a stored assembly (in the Spent Fuel Pool) without affecting the fuel category.

Figure 4.3.1-1
(continued)
Allowable Storage Arrays
See notes 1 - 7 below for use of Figure 4.3.1-1

Definition	Illustration		
Array A Category 6 assembly in every cell.		6	6
		6	6
Array B Category 3 assembly in 3-of-4 cells, with empty cell in the fourth cell.		3	3
		3	X
Array C Checkerboard pattern of diagonally-opposed Category 1 assemblies with empty cells.		1	X
		X	1
Array D Checkerboard pattern of two face-adjacent Category 5 assemblies with an empty cell and Category 1 assembly. Allows for transition from Array C and other arrays.		5	5
		1	X
Array E Checkerboard pattern of two diagonally-opposed Category 2 assemblies with an empty cell and Category 4 assembly.		4	2
		2	X
Array F Checkerboard pattern of diagonally-opposed Category 7 consolidated rod storage canisters and empty cells, which may be filled with assembly nozzles, guide tubes, and grids.		7	X
		X	7
Array G 3-by-3 pattern of Category 5 assemblies with an RCCA loaded in the center assembly.	5	5	5
	5	5R	5
	5	5	5

Figure 4.3.1-1
(continued)
Allowable Storage Arrays
See notes 1 - 7 below for use of Figure 4.3.1-1

Notes:

1. In all arrays, an assembly of higher Fuel Category number can replace an assembly designated with a lower Fuel Category number.
2. Category 1 is fuel up to 5.0 weight percent U-235 enrichment and does not credit burnup.
3. Fuel Categories 2 through 6 are determined from Tables 4.3.1-2 or 4.3.1-3.
4. An "R" designates a location that requires insertion of an RCCA in the fuel assembly.
5. An "X" designates a location that requires an empty cell, except that the empty cells in Array F may store assembly structural materials including nozzles, guide tubes, and grids.
6. An empty (water-filled) cell may be substituted for any fuel-containing cell in all storage arrays.
7. Array F shall only interface with Array A, and no other.

Enclosure 3

Marked-Up Technical Specification Bases Pages

15 pages follow

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

The spent fuel storage pool is a two compartment pool as described in Reference 1. These 2 compartments are referred to as Pool 1 and Pool 2. Pool 1 has up to 462 storage positions. Pool 2 has up to 1120 storage positions.

Either pool is designed to accommodate fuel of various initial enrichments (up to 5 weight percent (w/o)) which ~~have accumulated minimum burnups and decay times within the unrestricted domain in accordance with Figure 3.7.17-1 of Technical Specification (TS) 3.7.17, "Spent Fuel Pool Storage". Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with paragraph 4.3.1.1~~ in TS Section 4.3, "Fuel Storage".

satisfy the storage requirements described

TS Section 3.7.17, "Spent Fuel Pool Storage" and

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 1.0~~0~~ be evaluated in the absence of soluble boron. The double contingency principle discussed in Reference 2 and the April 1978 NRC letter (Ref. 3) allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Safe operation of the spent fuel pool may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.17, "Spent Fuel Pool Storage" and by maintaining boron concentration in accordance with this LCO.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The spent fuel pool criticality analysis (Ref. 4 ~~and 5~~) addresses all the fuel types currently stored in the spent fuel pool and in use in in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs, (both 0.400" and 0.422" O.D. designs) and the Exxon fuel assembly types in storage in the spent fuel pool.

Accident conditions which could increase the k_{eff} were evaluated including:

a. A new fuel assembly drop on the top of the racks;
b. A new fuel assembly misloaded between rack modules;
c. A new fuel assembly misloaded into an incorrect storage rack location;
d. Intramodule water gap reduction due to a seismic event; and
e. Spent fuel pool temperature greater than 150 °F.

For an occurrence of these postulated accident conditions, the double contingency principle of Reference 2 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the ~~464~~ 359 ppm required to maintain k_{eff} less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed (Ref. 4 ~~and 5~~) to determine the amount of soluble boron required to offset the highest reactivity increase caused by these postulated accidents and to maintain k_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of ~~730~~ 910 ppm (assuming a conservatively low boron-10 atom percent of 19.4) was adequate to mitigate these postulated criticality related accidents and to maintain k_{eff} less than or equal to 0.95. This specification ensures the spent fuel pool contains

BASES

APPLICABLE SAFETY ANALYSES (continued)

adequate dissolved boron to compensate for the increased reactivity caused by these accidents. ~~The 1800 ppm spent fuel pool boron concentration limit in this specification was chosen to be consistent with the original boron concentration limit required for a spent fuel cask containing fuel. The current boron concentration limit required for a spent fuel cask is controlled by the Prairie Island Independent Spent Fuel Storage Installation Technical Specifications.~~

A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event.

~~A spent fuel pool boron concentration of 750 ppm was required by the previous spent fuel rack criticality analysis to ensure that the spent fuel rack k_{eff} would be less than or equal to 0.95 for the allowable storage configurations, excluding accidents. Therefore the spent fuel pool dilution analysis utilized 750 ppm as the endpoint of the analysis to determine the dilution time and volume of water required to dilute the spent fuel pool from the 1800 ppm Technical Specification limit.~~ The current spent fuel rack criticality analysis (Ref. 4 ~~and 5~~) only requires a boron concentration of ~~464~~ 359 ppm (assuming a conservatively low boron-10 atom percent of 19.4) to ensure that the spent fuel rack k_{eff} will be less than or equal to 0.95 for the allowable storage configuration, excluding accidents. Therefore the spent fuel pool boron dilution analysis which assumes 750 ppm as the endpoint of the analysis is conservative with respect to the endpoint of ~~464~~ 359 ppm since a larger volume of water would be required, which would take more time to dilute the spent fuel pool to ~~464~~ 359 ppm.

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

BASES (continued)

LCO

The fuel storage pool boron concentration is required to be ≥ 1800 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 4 ~~and 5~~. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage~~,~~ movement within the fuelstorage pool~~, and for loading and unloading a spent fuel storage cask~~.

insert space

and

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 spent 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. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. This does not preclude movement of a fuel assembly to a safe position.

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.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

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

REFERENCES

1. USAR, Section 10.2.
2. ANSI/ANS-8.1-1983.
3. Nuclear Regulatory Commission, 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.
4. "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", WCAP-~~16517~~ 17400-NP, Revision 0, Westinghouse Electric Company, ~~November 2005~~ July 2011.
5. ~~Addendum 1 to WCAP-16517-NP, Revision 0, "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", Bishop, T.C., February 2008.~~

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel storage pool is a two compartment pool as described in the USAR (Ref. 1). These 2 compartments are referred to as Pool 1 and Pool 2.

Criticality considerations provide the primary basis for storage limitations.

Pool 1 may contain up to 462 storage positions, except when the pool is used for cask laydown. In the latter case, only 266 storage positions are available since 4 storage racks must be removed to accommodate the storage cask. Pool 2 has up to 1120 storage positions.

satisfy the storage requirements described in Specification 3.7.17 and Section 4.3, Fuel Storage.

Pools 1 and 2 are designed to accommodate fuel of various initial enrichments (up to 5 weight percent (w/o)), which ~~have accumulated minimum burnups and decay times within the unrestricted domain according to Figure 3.7.17-1 in the accompanying LCO.~~

~~Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.~~

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 1.0~~0~~ be evaluated in the absence of soluble boron. The double contingency principle discussed in Reference 2 and the April 1978 NRC letter (Ref. 3) allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel pool may therefore be achieved by controlling the location of each

the values required by 10 CFR 50.68(b)

BACKGROUND (continued)

assembly in accordance with the accompanying LCO and maintaining boron concentration in accordance with LCO 3.7.16.

APPLICABLE SAFETY ANALYSES

The hypothetical criticality accidents can only take place during or as a result of the movement of an assembly (Ref. 4 and 5). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Fuel Storage Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by verifying the appropriate checkerboarding after each fuel handling campaign, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for criticality accidents, the operation may be under the auspices of the accompanying LCO.

these criteria for

The spent fuel storage racks have been analyzed in accordance with the methodology contained in Reference 4. That methodology ensures that the spent fuel rack multiplication factor, k_{eff}, is less than ~~0.95 as recommended by ANSI 57.2-1983 (Ref. 6) and NRC guidance (Ref. 3)~~. The codes, methods and techniques contained in the methodology are used to satisfy ~~this criterion on~~ k_{eff}. The resulting Prairie Island spent fuel rack criticality analysis allows for the storage of fuel assemblies with enrichments up to a maximum of 5.0 (nominal 4.95% ± 0.05%) weight percent U-235 while maintaining k_{eff} ~~≤ 0.95 including uncertainties and credit for soluble boron. In addition, sub-criticality of the pool (k_{eff} < 1.0) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. Credit is taken for radioactive decay time of the spent fuel and for the presence of fuel rods containing gadolinium burnable poison.~~

< 1.0 (including uncertainties) if flooded with unborated water and Keff ≤ 0.95 (including uncertainties) with credit for soluble boron. The analysis determined that a minimum soluble boron concentration of 359 ppm (at a conservatively low boron-10 atom percent of 19.4) will ensure any loaded configuration Keff will be ≤ 0.95. In addition, the analysis differentiated a fuel assembly operated during Operating Cycle 1 - 4 from an assembly operated after Cycle 4 in determining the assembly's reactivity. Credit is taken for the radioactive decay time of the spent fuel. No credit is given for any gadolinium burnable poison in the fuel.

The criticality analysis (Ref. 4 ~~and 5~~) ~~utilized~~ the following storage configurations to ensure that the spent fuel pool will remain subcritical ~~during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment~~:

specifically analyzed each of

when fuel is placed in accordance with Section 4.3.1.1.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~a. The first storage configuration utilizes a pattern to accommodate new or low burnup fuel with maximum enrichment of 5.0 w/o U-235. This configuration stores "burned" and "fresh" fuel assemblies in a 3x3 checkerboard pattern as shown in Figure 4.3.1-1. Fuel assemblies stored in burned" cell locations are selected based on a combination of initial enrichment, discharge burnup and decay time (Figures 4.3.1-3 and 4.3.1-4). The criteria for the fuel stored in the "burned" locations is also dependent on the presence of rods containing gadolinium in the center "fresh" fuel assembly. The use of empty cells is also an acceptable option for the "fresh" and "burned" cell locations. This will allow the storage of new or low burnup fuel assemblies in the outer rows of the spent fuel storage racks because the area outside the racks can be considered to be empty cells.~~

~~Fuel assemblies that fall into the restricted range of Figure 3.7.17-1 are required to be stored in "fresh" cell locations as shown in Figure 4.3.1-1. The criteria included in Figure 3.7.17-1 for the selection of fuel assemblies to be stored in the "fresh" cell locations is based on a combination of initial enrichment, decay time and discharge burnup.~~

~~b. The second storage configuration does not utilize any special loading pattern. Fuel assemblies with burnup, initial enrichment and decay time which fall into the unrestricted range of Figure 3.7.17-1 can be stored anywhere in the region with no special placement restrictions.~~

~~The burned/fresh fuel checkerboard region can be positioned anywhere within the spent fuel racks, but the boundary between the checkerboard region and the unrestricted region must be either:~~

~~a. Separated by a vacant row of cells; or~~

Replace with text in "Insert A" (attached).

and operated at reactor power conditions including Extended Power Uprate (analyzed core power of 1811 MWt).

APPLICABLE SAFETY ANALYSES (continued)

~~b. The interface must be configured such that there is one row carryover of the pattern of burned assemblies from the checkerboard region into the first row of the unrestricted region (Figure 4.3.1-2).~~

Specification 3.7.17 and Section 4.3 ensure that fuel is stored in the spent fuel racks in accordance with the storage configurations assumed in the spent fuel rack criticality analysis (Ref. 4 ~~and 5~~).

The spent fuel pool criticality analysis addresses all the fuel types currently stored in the spent fuel pool and in use in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs (both 0.400" and 0.422" O.D. designs), and the Exxon fuel assembly types in storage in the spent fuel pool.

Accident conditions which could increase the k_{eff} were evaluated including:

a. A new fuel assembly drop on the top of the racks;

b. A new fuel assembly misloaded between rack modules;

c. A new fuel assembly misloaded into an incorrect storage rack location;

d. Intramodule water gap reduction due to a seismic event; and

e. Spent fuel pool temperature greater than 150°F.

359

For an occurrence of these postulated accident conditions, the double contingency principle of Reference 2 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the ~~464~~ ppm required to maintain k_{eff} less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its

(assuming a conservatively low boron-10 atom percent of 19.4)

BASES

APPLICABLE SAFETY ANALYSES (continued)

presence would be a second unlikely event.

910

Westinghouse Electric Company LLC calculations (Ref. 4) were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by these postulated accidents and to maintain k_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of ~~730~~ ppm was adequate to mitigate these postulated criticality related accidents and to maintain k_{eff} less than or equal to 0.95.

This value was selected to provide a nominal margin above the calculated limiting value of 359 ppm.

Specification 3.7.16 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by a mispositioned fuel assembly or a loss of spent fuel pool cooling.

Section 4.3 requires that the spent fuel rack k_{eff} be less than or equal to 0.95 when flooded with water borated to ~~730~~ ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that ~~an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm (thus conservative with respect to the endpoint of 730 ppm) is not a credible event.~~

400

When the requirements of Specification 3.7.17 are not met, immediate action must be taken to move any noncomplying fuel assembly to an acceptable location to preserve the double contingency principle assumption of the criticality accident analysis.

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

sufficient time would be available for operators to recognize and terminate a dilution event that started at the spent fuel pool boron concentration of 1800 ppm and terminated at 750 ppm; providing significant margin to the 400 ppm value provided in Section 4.3.1.1

Section 4.3.1.1 and

BASES (continued)

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with ~~Figure 3.7.17-1 in~~ the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.95, with credit given for boron in the water.

1.0 in unborated water and ≤

~~Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.~~

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.

Section 4.3.1.1

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with ~~Figure 3.7.17-1 or Specification 4.3.1.1~~, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with ~~Figure 3.7.17-1 or Specification 4.3.1.1~~.

Section 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 (continued)

fuel assembly, fuel insert, or other hardware is placed in the storage racks in accordance with Section 4.3.1.1 and

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the ~~initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 in~~ the accompanying LCO. ~~For fuel assemblies in the restricted range of Figure 3.7.17-1 performance of this SR will ensure compliance with Specification 4.3.1.1.~~

The Frequency of this SR is prior to storing or moving a fuel assembly.

and other hardware affected by a fuel handling campaign are placed

SR 3.7.17.2

This SR verifies that the fuel ~~assemblies~~ in the spent fuel storage racks are stored in accordance with the requirements of LCO 3.7.17 and Section 4.3.1.1.

The intent of the SR is to ensure that the storage configuration following a spent fuel pool campaign was completed accurately. This SR helps ensure that storage arrays affected by the campaign will continue to meet subcriticality criteria of TS 4.3.1.1.

~~The intent of this SR is to not require completion of the spent fuel pool inventory verification during interruptions in fuel handling during a defined fuel handling campaign. No spent fuel pool inventory verification is required following fuel movements where no fuel assemblies are relocated to different spent fuel rack locations.~~

The Frequency of this SR requires performance within 7 days after the completion of any fuel handling campaign which involves:

a. The relocation of fuel assemblies within the spent fuel pool; ~~or~~

b. The addition of fuel assemblies to the spent fuel pool

; or

c. The relocation of non-fuel materials in the spent fuel storage racks. Such materials include rod control cluster assemblies (RCCAs), failed fuel baskets, neutron source assemblies, and metal waste materials.

The extent of a fuel handling campaign will be defined by plant administrative procedures. Examples of a fuel handling campaign would include all the fuel handling performed during a refueling outage or associated with the placement of new fuel into the spent fuel pool.

The frequency of this SR is based on providing timely verification without imposing interruption to the fuel handling processes during a defined fuel handling campaign.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.2 (continued)

The 7 day allowance for completion of this SR provides adequate time for completion of the spent fuel pool inventory verification while minimizing the time a fuel assembly may be misloaded in the spent fuel pool. If a fuel assembly is misloaded during the fuel handling campaign, the minimum boron concentration required by LCO 3.7.16 will ensure that the spent fuel rack k_{eff} remains within limits until the spent fuel inventory verification is performed

insert a period.

REFERENCES

1. USAR, Section 10.2.

2. ANSI/ANS-8.1-1983.

3. Nuclear Regulatory Commission, 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.

17400

4. "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", WCAP-~~16517~~-NP, Revision 0, Westinghouse Electric Company, ~~November 2005~~

July 2011

5. ~~Addendum 1 to WCAP-16517-NP, Revision 0, "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", Bishop, T.C., February 2008~~

Unused.

6. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983.

In each array described below, the fuel assemblies are first categorized by their relative reactivity by using Tables 4.3.1-1 through 4.3.1-3. The purpose of any array is to combine the use of low-reactivity fuel and/or empty cells to offset higher-reactivity fuel. Parameters that define a fuel category include initial enrichment, burnup, decay time, and the fuel assembly's operating cycles. Each of the approved arrays and the array interface requirements are described below:

Array A represents a 2x2 "all-cell" configuration of low-reactivity (Category 6 fuel) in every cell. Category 6 assemblies are those that satisfy the burnup requirement (for Category 6) defined by the polynomial equations provided in Tables 4.3.1-2 or 4.3.1-3.

Array B represents a 2x2 cell configuration of high-reactivity (Category 3 fuel) in three cells, offset by the empty cell in the fourth location. This array was provided to accommodate the temporary placement of high-reactivity (e.g., once-burned) fuel during a refueling or maintenance outage.

Array C represents a 2x2 cell configuration of new fresh fuel (Category 1 fuel), offset by the empty cells in the checkerboard pattern to reduce the overall reactivity of the configuration. This array was provided to accommodate the temporary pre-staging of fresh fuel for a refueling outage or to accommodate low-burnup fuel that might be discharged prior to a full cycle of depletion (maintenance outage or fuel failure).

Array D represents a 2x2 cell configuration of one fresh fuel assembly plus two medium-reactivity fuel (Category 5 fuel) assemblies offset by one empty cell face-adjacent to the fresh assembly. This array was provided to accommodate more efficient storage for a fresh (or low-burnup) assembly and other medium-burnup fuel that might be required for a mid-cycle maintenance outage requiring core offload. For any given amount of low-burnup (Category 1) fuel that an outage may require, this configuration would require approximately half as many empty cells as would otherwise be required if Array C were used.

Array E represents a 2x2 cell configuration of two low-burnup fuel assemblies (Category 2) in a checkerboard pattern with one medium burnup (Category 4) assembly plus an empty cell in a checkerboard pattern to offset the reactivity. This array provides efficient storage of once-burned and twice-burned fuel that would be discharged for a refueling outage.

Array F represents a 2x2 cell configuration that is specifically provided to accommodate the fuel rods from 36 fuel assemblies that were consolidated into 18 Consolidated Rod Storage Canisters (CRSCs) and loaded in a checkerboard pattern with the remnants of the 36 fuel assemblies. This array was specifically analyzed with significant stainless steel materials in the cells face-adjacent to the CRSCs to address the remnants as described in the USAR Section 10.2.1.5. Otherwise, as described in TS Figure 4.3.1-1 Note 5, cells designated to be empty in the other arrays must be empty.

Array G represents a 3x3 cell configuration very similar to Array A "all-cell", but requiring a lower burnup value for the fuel (Array A requires Category 6 fuel whereas Array F only requires Category 5 fuel). This relaxation is offset by the requirement to insert and maintain a Rod Control Cluster Assembly (RCCA) in the center location to reduce the overall reactivity of the modified "all-cell" configuration. This array was provided for spent (thrice-burned) fuel that may not have accumulated enough burnup to qualify as Category 6 fuel. This array also puts into use RCCAs that would otherwise just take up space in the spent fuel pool.

Rack interface requirements: The Technical Specifications do not provide any unique rules for the interface between rack modules because all the racks in the SFP have identical fuel cell design and the actual physical gap between rack modules is ignored in the analysis (i.e., there is no credit taken for the gaps between rack modules).

Array interface requirements: Technical Specifications provide only one special interface requirement between different arrays. This specific interface is described in Figure 4.3.1-1 Note 7 (Array F shall interface only with Array A) and was specifically analyzed. Otherwise, the Technical Specifications do not provide any unique rules for the interface between arrays. Rather, the Technical Specifications require that all fuel in the spent fuel pool satisfy one of the required arrays, even in transitions between two major arrays.

Enclosure 5

Westinghouse Affidavit

7 pages follow

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: NSP-11-112

CAW-11-3211

July 18, 2011

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-17400-P, Revision 0, "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-11-3211 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Xcel Energy.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-11-3211, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

W. J. Arndt /For
J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared B. F. Maurer, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

B. F. Maurer, Manager
ABWR Licensing

Sworn to and subscribed before me
this 18th day of July 2011

Cynthia Olesky
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Cynthia Olesky, Notary Public
Manor Boro, Westmoreland County
My Commission Expires July 16, 2014
Member, Pennsylvania Association of Notaries

(1) I am Manager, ABWR Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(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 the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, 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.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

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.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-17400-P, Revision 0, "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary), dated July 2011, for submittal to the Commission, being transmitted by Xcel Energy letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Demonstrate the sub-criticality of the spent fuel pool.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for the purpose of demonstrating the sub-criticality of the spent fuel pool.

(b) Westinghouse can sell support and defense of spent fuel pool criticality safety analysis.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

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 calculations 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.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or 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 (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Enclosure 7

List of Regulatory Commitments

1. In conjunction with implementation of the proposed Technical Specifications (TS), procedures will be revised to require positive controls for the movement of all fissile material in the Spent Fuel Pool (including Consolidated Rod Storage Canister (CRSC), Fuel Rod Storage Canister (FRSC), and Failed Fuel Pin Basket (FFPB)) and all non-fuel material (including fuel assembly cages and dummy fuel assemblies) placed into a storage rack. These positive controls will be comparable to those used for fuel assemblies (e.g., Fuel Transfer Log). The procedure revisions will include a post-campaign validation of all the affected Spent Fuel Pool (SFP) locations, whether fresh fuel is involved in the campaign or not.

2. In conjunction with implementation of the proposed TS, procedures will be revised to require an assessment of a fuel assembly's cumulative exposure to rodded power operation in the core prior to moving that fuel assembly into the SFP storage racks. If an assembly experiences more than 1 gigawatt day per metric ton uranium (GWd/MTU) of core average rodded operation, the assembly shall either be treated as Fuel Category 1 or evaluated to determine which Fuel Category is appropriate for safe storage of the assembly.

3. In conjunction with implementation of the proposed TS, an assessment of existing spent fuel inventory will be performed to ensure cumulative exposure to rodded operation for any assembly does not exceed 1 GWd/MTU. If an assembly experienced more than 1 GWd/MTU of core average rodded operation, the assembly shall either be treated as Fuel Category 1 or evaluated to determine which Fuel Category is appropriate for safe storage of the assembly.