SEP 16 2005

L-PI-05-075
10 CFR 50.90

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
License Nos. DPR-42 and DPR-60

Supplement to License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage" (TAC Nos. MC5811 and MC5812)

By letter dated February 1, 2005, Nuclear Management Company (NMC) submitted an LAR to revise the spent fuel pool criticality analyses and Technical Specifications (TS) 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage". This letter supplements the subject LAR. NMC submits this supplement in accordance with the provisions of 10 CFR 50.90.

By letter dated July 26, 2005, the NRC Staff requested additional information related to the subject LAR. Enclosure 1 to this letter states the NRC Staff questions and the NMC responses. Enclosure 2 provides a markup of TS Figure 4.3.1-1 with additional changes in support of the response to question 10. Enclosure 3 provides the clean revised TS Figure 4.3.1-1 retyped. The TS Figure 4.3.1-1 provided in this supplement supersedes the TS Figure 4.3.1-1 provided with the original submittal. Enclosure 4 provides Exhibit A of the February 1, 2005 submittal revised to support the response to Question 1.

The proposed changes in this supplement do not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the February 1, 2005 submittal and February 22, 2005 supplement.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR supplement by transmitting a copy of this letter and enclosures to the designated State Official.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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

SEP 16 2005

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

Enclosures (4)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Response to Request for Additional Information

Nuclear Regulatory Commission Staff Question 1:

In its amendment request, NMC provided a brief synopsis of the licensing basis for the Spent Fuel Pool (SFP) criticality analyses. The acceptance criteria cited by NMC in Section 5.2, "Applicable Regulatory Requirements/Criteria" are codified in NRC regulations. Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.68, "Criticality accident requirements," provides NRC acceptance criteria for the safe storage of fuel in the spent fuel pool. Since NMC has proposed to take advantage of the regulatory advantages afforded by 10 CFR 50.68, the approval of NMC's amendment request will necessitate a satisfactory demonstration of compliance with all of the 10 CFR 50.68 acceptance criteria. This was not provided in the amendment request. The NRC cannot approve a partial implementation of 10 CFR 50.68. Therefore, the staff requests that the licensee provide a summary of how each of the eight criteria in 10 CFR 50.68(b) will be met in the PINGP spent fuel pools.

Nuclear Management Company (NMC) Response:

NMC did not address compliance with each of the Title 10 Code of Federal Regulations 50.68 (10 CFR 50.68) acceptance criteria because it does not apply to the Prairie Island Nuclear Generating Plant (PINGP). 10 CFR 50.68 states that each licensee "shall comply with <u>either 10 CFR 70.24</u> of this chapter <u>or</u> the requirements in paragraph (b) of this section." (emphasis added) The PINGP licensing basis for spent fuel pool criticality is based on compliance with 10 CFR 70.24 and therefore, 10 CFR 50.68 does not apply to PINGP. To clarify this issue, this LAR supplement withdraws the discussion entitled, "10 CFR 50.68, Criticality Accident Requirements" from page 10 of Exhibit A. Revision 1 of Exhibit A of the LAR dated February 1, 2005, with the discussion of 10 CFR 50.68 removed from page 10, is provided in Enclosure 4 to this letter.

Nuclear Regulatory Commission Staff Question 2:

In Section 1.2, NMC stated that it modeled the unborated moderator (water) with a density equal to 1.0 g/cc. The staff agrees that the assumption of full density moderator is conservative if the moderator temperature coefficient (MTC) is negative under nominal storage conditions in the spent fuel pool. However, Tables 3-4, 3-5, and 3-6 include a pool temperature bias that appears to indicate

that full density water does not provide optimum moderating conditions. NRC regulations (10 CFR 50.68) and guidance documents require that the criticality analyses be performed under optimum moderation conditions. Since under some design configurations, the MTC can be positive, the staff requests the licensee describe what analyses it performed to demonstrate that the MTC under the most limiting storage conditions in the spent fuel pool was negative and that the full density moderator assumption was conservative. Additionally, if a bias is appropriate, the staff requests that the licensee justify the use of a bias based on previous criticality analyses that were dependent of different fuel storage conditions.

NMC Response:

The PINGP analyses have followed spent fuel criticality analysis guidance documents which require criticality analyses to be performed under optimum moderating conditions. (As noted in response to Question 1 above, 10 CFR 50.68 does not apply to PINGP.) The temperature bias was recalculated with KENO for the "All-Cell" storage configuration and the resulting value is 0.00635. The temperature bias given in this LAR is 0.00640 Δ k_{eff} units. The temperature bias calculated in the previous analysis was for the same storage configuration ("All-Cell") and is the maximum value for all three storage configurations. Since the temperature bias covers the entire spent fuel pool temperature range, the analysis results are based upon the "optimum" water density.

Nuclear Regulatory Commission Staff Question 3:

In Section 2.2, NMC described the storage modules in the PINGP spent fuel pools. The licensee stated that, "The modules are separated by a minimum water gap of 1 inch." Since the spacing between fuel assemblies is a key parameter in the analysis of the maximum k_{eff} between spent fuel storage modules, the staff requests that the licensee describe how the minimum water gap is assured.

NMC Response:

The spent fuel racks at Prairie Island were installed using contractor procedures AZSF-17-PI and AZSF-18-PI. As part of these procedures, four gapping tools were attached to each storage module. Each module was installed with the assistance of an underwater diver such that the gapping tools were in contact with the surrounding walls and/or storage modules, as appropriate.

Nuclear Regulatory Commission Staff Question 4:

In Section 3.1, NMC stated that scoping calculations were performed for the ^{235}U loading and storage configurations considered in the amendment request to determine the most reactive fresh fuel assembly design. However, the licensee did not provide the results for these scoping calculations. Since the proper selection of the design basis fuel assembly is essential for ensuring the maximum k_{eff} is calculated and NRC regulations are satisfied, the staff requests that NMC provide a table of the results of the scoping calculations that supports its determination of the most reactive fresh fuel assemblies under the different storage configurations proposed in the amendment.

NMC response:

The following table provides the results of scoping calculations employed to determine the most reactive fresh fuel assembly type for the 3x3 storage configuration. These results demonstrate that a fresh Optimized Fuel Assembly (OFA) assembly is more reactive than a fresh Standard assembly for the 3x3 storage configuration.

Enrichment (w/o)	Assembly Burnup (MWD/MTU)	k_{eff} by Fuel Type	
		OFA	Standard
3.0	25,000	0.98183	0.97751
5.0	55,000	0.96383	0.95807

Nuclear Regulatory Commission Staff Question 5:

In Section 3.3, NMC stated the following: "The [fuel and moderator temperature] values are based on mid-cycle temperature profiles for Prairie Island Units 1 and 2." The proper selection of fuel and moderator temperatures as well as soluble boron concentrations is critical in the determination of a realistically conservative depletion analysis. Therefore, the staff requests that NMC provide a comparison of the data used in the depletion analyses to historical operating conditions at PINGP. The licensee must demonstrate that the assumptions used in its depletion analysis conservatively bound the historical operating conditions at PINGP.

NMC response:

The following parameters were employed for the Discrete Integral Transport (DIT)

complete accounting of all tolerances and their associated reactivity effects. Therefore, the staff requests that NMC provide an analysis of the other tolerances not considered in its amendment request to ensure that the k_{eff} will remain below NRC regulatory limits.

NMC response:

Westinghouse has utilized essentially the same set of tolerances for a number of utility license amendment applications. The NRC has approved this approach, with the same set of tolerances, in plant submittals which include: R. E. Ginna Nuclear Power Plant – Amendment RE: Revision to the Storage Configuration Requirements Within the Existing Storage Racks and Taking Credit for a Limited Amount of Soluble Boron (TAC No. MA8443) dated December 7, 2000; Millstone Power Station, Unit No. 2 – Issuance of Amendment RE: Spent Fuel Pool Requirements (TAC No. MB 3386) dated April 1, 2003; Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment RE: Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis (TAC Nos. MB2982 and MB2984) dated September 2002; and Joseph M. Farley Nuclear Plant, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MC6987 and MC6988) dated June 28, 2005.

Nuclear Regulatory Commission Staff Question 7:

Additionally, in Section 3.4, NMC stated that the tolerance analyzed for the gadolinia concentration is equal to -0.2 weight percent. However, NMC did not provide a basis for the uncertainty assumed in the analysis. The staff requests that NMC provide a technical basis for the uncertainty assumed and a justification for why this uncertainty provides an appropriately conservative result.

NMC response:

The actual manufacturing tolerance for gadolinia concentration is $\pm$ 3%. Based upon a nominal 4.0 weight percent (w/o) gadolinia (utilized for the fresh fuel assembly containing 4 gadolinia rods) the maximum deviation in gadolinia concentration would be 0.12 w/o. The gadolinia tolerance employed for this analysis, 0.20 w/o, is therefore conservative.

Nuclear Regulatory Commission Staff Question 8:

In Section 3.5, NMC provided a description of the cooling (decay) time credit employed in the criticality analyses. NMC determined cooling time credits on discrete 5-year intervals. Since appropriately classifying assemblies based on

cooling time will be essential for ensuring subcriticality margins are maintained, the staff requests that the licensee describe how it will conservatively apply the cooling time credit to assemblies that fall between the discrete intervals calculated (e.g., assemblies with 7.5 or 12.5 years of cooling time).

NMC response:

To determine the required burnup time for fuel assemblies that fall between the calculated discrete cooling time intervals, an interpolation program is used. This interpolation introduces a slight conservatism into the process. Because the discrete cooling time curves are not linear, using a linear interpolation results in a slightly more restrictive definition of which assemblies are acceptable for storage than is required by the curves, that is, for the same enrichment and cooling time, a slightly greater assembly burnup is required to be acceptable for storage.

Nuclear Regulatory Commission Staff Question 9:

In Section 3.1.2, NMC provides a list of four assumptions that were used to represent the gadolinium in the fresh fuel pellets in the KENO V.a model of the 3x3 storage region. However, the licensee did not provide a basis describing how each of these assumptions will provide a conservative representation of fresh fuel assemblies at PINGP. Therefore, the staff requests that the licensee provide a technical justification demonstrating that each of the assumptions provides conservative margin in the criticality analyses.

NMC response:

The four assumptions listed in Section 3.1.2 concerning the modeling of gadolinia shimmed fuel rods and an explanation for each are given below:

1) A six inch burnable absorber "cutback" is employed at the top and bottom of the fuel rod. The "cutback" is defined as a section of the fuel rod which does not contain gadolinia.

   This assumption was employed to conservatively model the sections of the fuel rod which do not contain gadolinia. The "cutback" size is therefore limited to less than or equal to six inches on the top and bottom of fuel rods which contain gadolinia. The design specified cutback size is less than 6 inches.

2) The Gd_2O_3 amount is limited to four fuel pins at a concentration of 4.0 w/o Gd_2O_3.

   These assumptions were made to minimize the reactivity hold-down of gadolinia in shimmed fuel assemblies. In the PINGP use of the TS figures, any fuel assemblies which contain less than 4 gadolina shimmed rods or less than 4.0 w/o Gd_2O_3 must be treated as fresh unshimmed fuel assemblies (see answer to question 10 below).

3) The ^{235}U enrichment is reduced to 4.0 w/o ^{235}U in the shimmed portion of the fuel rod for fuel temperature considerations.

By manufacturing procedure, the enrichment of the shimmed portion of the fuel rod is less than the enrichment of the unshimmed fuel rods by employing the derivative 5 % enrichment reduction per unit weight per cent of Gadolinia. Since the shimmed rods are modeled with 4.0 w/o Gd_2O_3, the necessary enrichment reduction is 20 %.

4) The ^{235}U enrichment in the blanket region of the shimmed fuel rods is also reduced to 4.0 w/o ^{235}U .

Blanket regions of fuel rods have an enrichment much less than 4.0 w/o ^{235}U . However, in this analysis blanketed fuel assemblies are conservatively not modeled. Therefore, the entire length of fuel rods which contain gadolinia is modeled with an enrichment equal to 4.0 w/o ^{235}U. The actual enrichment is less than this value.

Nuclear Regulatory Commission Staff Question 10:

NMC's proposed TS Figure 4.3.1-1 allows the storage of fresh fuel assemblies in the spent fuel pool with or without gadolinium based on ensuring that adjacent spent fuel assemblies satisfy minimum burnup requirements. However, the licensee did not propose TS limits that will require a minimum gadolinium loading, in accordance with assumptions used in the criticality analyses, in the fresh fuel prior to placing it in the designated storage locations. Therefore, the staff requests that the licensee provide additional information demonstrating that sufficient controls will be put in place to ensure fresh fuel assemblies loaded in the spent fuel storage racks will be appropriately controlled based on the amount of gadolinium.

NMC response:

The legend for "Fresh Fuel" on proposed TS Figure 4.3.1-1 is revised to include minimum gadolinia (GAD) requirements. The revised figure is provided in Enclosures 2 and 3 of this supplement. With this change, the minimum required gadolinia loading in the fresh fuel is specified and will be controlled by NMC procedures which reference this Figure.

Nuclear Regulatory Commission Staff Question 11:

NMC's proposed Surveillance Requirement (SR) 3.7.17.1 requires that prior to storing or moving a fuel assembly in the spent fuel pool the licensee must "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 Specification 4.3.1.1." Although NMC is not proposing to change the wording of the SR, the proposed changes to the limiting condition for operation (LCO) Figures referenced in the SR necessitates a reevaluation of the SR effectiveness for ensuring proper storage of fuel assemblies. The licensee did not provide in its amendment request a description of the administrative process it will use to verify the parameters that govern fuel assembly storage requirements. Since the licensee intends to rely on administrative controls for prevention of accidents such as misloading of one or more fuel assemblies, the staff requests that the licensee provide a description of the controls to be implemented and a summary of how they are designed to minimize the potential for accidents that could challenge NRC's regulatory limits that are designed to prevent an inadvertent criticality.

NMC response:

This response is in accordance with clarification to the RAI provided by the NRC Staff in a telephone call with NMC Staff on August 24, 2005.

This LAR proposes an administrative change to Surveillance Requirement (SR) 3.7.17.1 which deletes reference to TS Figure 3.17-2 which will not exist when this LAR is approved. SR 3.7.17.1 requires verification by administrative means that the TS requirements of TS 3.7.17 and 4.3.1.1 are met. PINGP has successfully applied administrative controls on TS required spent fuel storage locations since soluble boron credit was granted by a license amendment issued June 12, 1997.

The NMC procedure which implements SR 3.7.17.1 assures that the TS requirements are met through the following documented checks prior to movement of a fuel assembly:

- Fuel assemblies to be moved are classified, based on TS 3.7.17 and 4.3.1.1 requirements, for the type of storage configuration and the classifications of fuel with which it is compatible for storage.
- Each of the fuel assemblies in the surrounding 24 fuel storage locations is classified, based on TS 3.7.17 and 4.3.1.1 requirements, and verified to be compatible with the fuel assembly to be moved. Guidance is provided for empty locations or storage locations that do not exist, for example, the center location under consideration is near the edge of the SFP.

Fuel assembly moves are not initiated unless the boron concentration requirements of TS 3.7.16 are met.

NMC has reviewed these procedures to evaluate the impact of this LAR. Administrative changes to TS figure references and fuel types are required; no changes to the procedure verifications and controls are required. The criticality analysis acceptance criteria which define the TS allowable storage configurations have not changed in this proposed LAR. The calculated margin to safety for the proposed TS allowable storage configurations remains the same or increased due to the new criticality analysis proposed in this LAR. Approval of this LAR will reduce the complexity of the TS fuel storage requirements. The analyses presented in this LAR demonstrate that the boron concentration required to mitigate a misloaded fuel assembly is far less than the TS required boron concentration. Thus a misloaded fuel assembly will not result in a criticality accident. Thus the proposed change with respect to the SR is not technically complex and the risk significance of the issue is very low. Based on these considerations, NMC determined that the existing plant procedural controls will continue to assure plant safety through compliance with TS requirements.

Nuclear Regulatory Commission Staff Question 12:

NMC's proposed TS Figure 3.7.17-1 provides minimum burnup versus enrichment curves for spent fuel storage in the pool. Proposed TS LCO 3.7.17 requires that assemblies that do not satisfy the TS Figure 3.7.17-1 combination of initial enrichment, burnup, and decay time limits for unrestricted storage must be stored in accordance with TS 4.3.1.1. However, the burnup versus enrichment curves provided in TS Figures 4.3.1-3 and 4.3.1-4 require higher burnups for the same initial enrichment and cooling times. Therefore, a spent fuel assembly that does not satisfy the unrestricted storage requirement of TS Figure 3.7.17-1 will not satisfy the acceptability requirements of either TS Figures 4.3.1-3 or 4.3.1-4. Based on this limitation, the staff believes that any assembly that does not satisfy the minimum burnup requirements of TS Figure 3.7.17-1 must be classified as a fresh fuel assembly and stored in accordance with fresh fuel loading configuration provided in TS Figure 4.3.1-1. The staff requests that the licensee confirm that these "restricted" spent fuel assemblies will be stored in accordance with fresh fuel assembly limitations and configurations.

NMC response:

The staff understanding of the TS Figures is correct. Assemblies that do not meet the TS Figure 3.7.17-1 restrictions on initial enrichment, burnup, and decay time are classified as a fresh fuel assembly and must be stored in accordance with the fresh fuel restrictions provided in TS Figure 4.3.1-1.

Nuclear Regulatory Commission Staff Question 13:

In addition to classifying TS Figure 3.7.17-1 "restricted" spent fuel assemblies as fresh fuel assemblies, low-burnup assemblies (e.g., those that may not have completed a full cycle of irradiation) that initially contained burnable poisons such as gadolinium may have higher residual reactivities than fresh fuel. The staff requests that NMC identify whether this limiting condition was considered in its criticality analyses. If the condition was not considered, the staff requests that NMC describe how low-burnup assemblies will be stored in the PINGP spent fuel pools.

NCM response:

The maximum reactivity of fuel assemblies containing gadolinia was considered. Westinghouse determined that the reactivity of gadolinia shimmed fuel assemblies (with 4 Gd_2O_3-UO_2 fuel rods at a concentration of 4.0 w/o Gd_2O_3), at any burnup greater than zero, is always lower than the initial reactivity at zero burnup.

Nuclear Regulatory Commission Staff Question 14:

In its amendment request, NMC included a reactivity depletion uncertainty in the calculation of the minimum soluble boron concentration requirement. This uncertainty was equal to 1.0 percent Δk_{eff} per 30,000 MWD/MTU of credited assembly burnup. However, it does not appear that a similar uncertainty was incorporated into the unborated maximum k_{eff} analyses (Tables 3-4, 3-5, and 3-6). Section 5.A.5.d of the August 19, 1998, NRC guidance document, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," states the following: "In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." The licensee did include a 5 percent uncertainty in the maximum burnup credited based on the MWD/MTU of burnup; however, this is not necessarily the equivalent of the 5 percent reactivity decrement described in the NRC guidance document. The staff's guidance on the inclusion of a 5 percent reactivity decrement is independent of whether the criticality analysis is being performed for borated or unborated conditions. Therefore, the staff requests that the licensee provide additional technical justification for not including a reactivity decrement in accordance with NRC guidance documents.

NMC response:

This LAR demonstrated that two acceptance criteria were met: 1) k_{eff} is less than or equal to 0.95 with soluble boron credit; and 2) k_{eff} is less than or equal to 1.0 when the spent fuel pool is unborated. The NRC guidance document, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," has been interpreted by Westinghouse to require a 5% burnup uncertainty and a reactivity depletion uncertainty equal to 1.0 % Δ k_{eff} per 30,000 MWD/MTU to be applicable only to acceptance criteria (1), that is, that k_{eff} be less than or equal to 0.95 with soluble boron credit. Several years ago Westinghouse began adding the 5% burnup uncertainty to both k_{eff} limits (unborated and borated) at the request of a customer. However, Westinghouse's opinion is that the requirement of applying a 5% burnup uncertainty or a reactivity depletion uncertainty is not required for the unborated k_{eff} limit. Westinghouse has not previously been asked to include the reactivity depletion uncertainty in establishing the unborated k_{eff} limit.

The NRC has previously approved this approach for the following plant submittals: R. E. Ginna Nuclear Power Plant – Amendment RE: Revision to the Storage Configuration Requirements Within the Existing Storage Racks and Taking Credit for a Limited Amount of Soluble Boron (TAC No. MA8443) dated December 7, 2000; Millstone Power Station, Unit No. 2 – Issuance of Amendment RE: Spent Fuel Pool Requirements (TAC No. MB 3386) dated April 1, 2003; Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment RE: Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis (TAC Nos. MB2982 and MB2984) dated September 2002; and Joseph M. Farley Nuclear Plant, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MC6987 and MC6988) dated June 28, 2005.

Nuclear Regulatory Commission Staff Question 15:

A major component of NMC's proposed changes to the SFP TSs is a reduction in the number of burnup versus enrichment curves that will govern fuel storage configurations. The current TSs delineate storage first based on the type of fuel assembly (e.g., Westinghouse Standard, Optimized, etc.), then on the presence and quantity of gadolinium rods, and finally on the burnup as a function of enrichment. The proposed TSs eliminate the first step of classifying based on fuel assembly type. Instead, NMC has chosen a more bounding analysis approach that identified the limiting fuel assembly and subsequently developed limiting burnup versus enrichment curves. It is reasonable to conclude that this bounding approach will require higher burnup limits to ensure subcritical storage configurations are established. However, in comparing the current TSs figures for fuel assembly burnup verses enrichment curves to those in the proposed TSs figures, it does not appear that the new figures are indeed bounding. For example, current TS Figure 3.7.17-2 provides burnup limits for Westinghouse Standard fuel assemblies for the "All Cell" configuration. In its new criticality analyses, NMC identified the Westinghouse Standard fuel assembly design as the most limiting in the "All Cell" configuration. However, the proposed TS Figure 3.7.17-1 that will govern loading of any assembly type into the "All Cell" configuration requires lower burnups, at given enrichments, than the current TS Figure 3.7.17-2. Similar differences exist between the proposed TS Figures 4.3.1-

3 and 4.3.1-4 and the corresponding current TS figures. The staff requests that the license provide a technical justification explaining any differences between the current and new criticality analyses that support the reduced burnup limits proposed.

NMC response:

The previous analysis relied heavily on a two dimensional (2D) methodology. The previous Phoenix calculations were performed for "reactivity equivalence" purposes. As already noted, the previous methodology has been replaced with an explicit three dimensional (3D) methodology.

Most of the differences between the new and old TS loading curves can be explained as follows:

- The old methodology conservatively reduced the gadolinia concentration in the shimmed section of the fuel rod by the factor 132/144 to account for axial cutback. The new methodology models the axial zoning of the gadolinia more explicitly.
- The old methodology reduced the gadolinia concentration further by the factors 0.9 and 0.97. The 0.9 factor was employed to account for "modeling" uncertainties and the 0.97 factor was employed to account for gadolinia concentration uncertainty. Neither of these factors was employed in the new methodology, except for the 3% gadolinia concentration tolerance in the uncertainty analysis. These overly conservative assumptions reduced the negative reactivity worth of the gadolinia.
- The previous methodology modeled assemblies with up to 20 gadolinia rods per assembly. Most of the "heavily shimmed" assembly designs have reactivity depletion effects which reduce the apparent worth of the gadolinia in fresh fuel assemblies. Therefore, the new TS loading curves are based upon more realistic and accurate modeling of a minimum number of gadolinia shimmed fuel rods in a fresh fuel assembly.

The previous and new "All-Cell" storage configuration results are very similar due to the fact that the 2D to 3D axial bias is very small (almost zero) for fuel assemblies with an initial enrichment up to 4.0 w/o ^{235}U. Above 4.0 w/o ^{235}U, the 2D to 3D axial bias is slightly positive. However, a comparison of the old and new analysis results indicates that the burnup requirements are very similar. Slight differences in the calculated biases and uncertainties, along with a more detailed modeling approach employed for the new analysis, have produced burnup requirements which are very similar to the old analysis results.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION (markup)

Technical Specification Page

4.0-5

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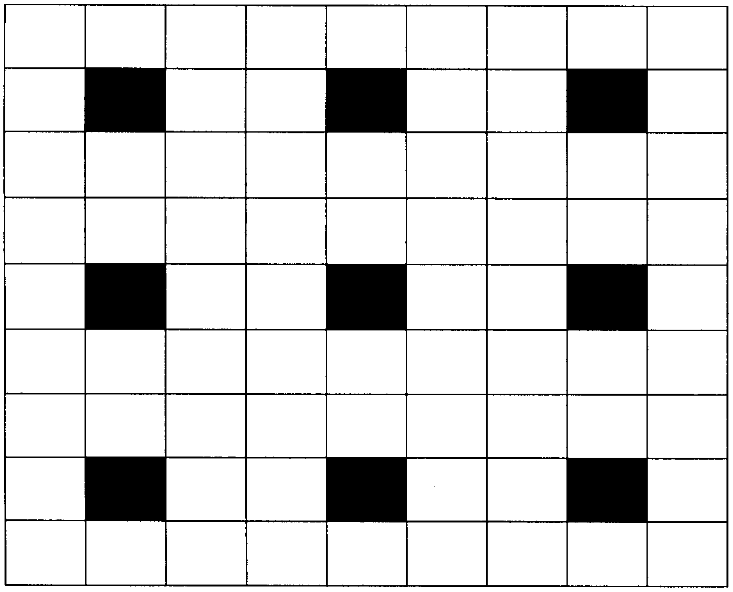

Fresh Fuel: Must be less than or equal to Nominal 4.95 w/o ^{235}U
No restrictions on burnup
Assemblies with GAD shall have a minimum of 4 fuel rods with a minimum concentration of 4.0 w/o Gd_2O_3

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

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

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ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION (retyped)

Technical Specification Page

4.0-5

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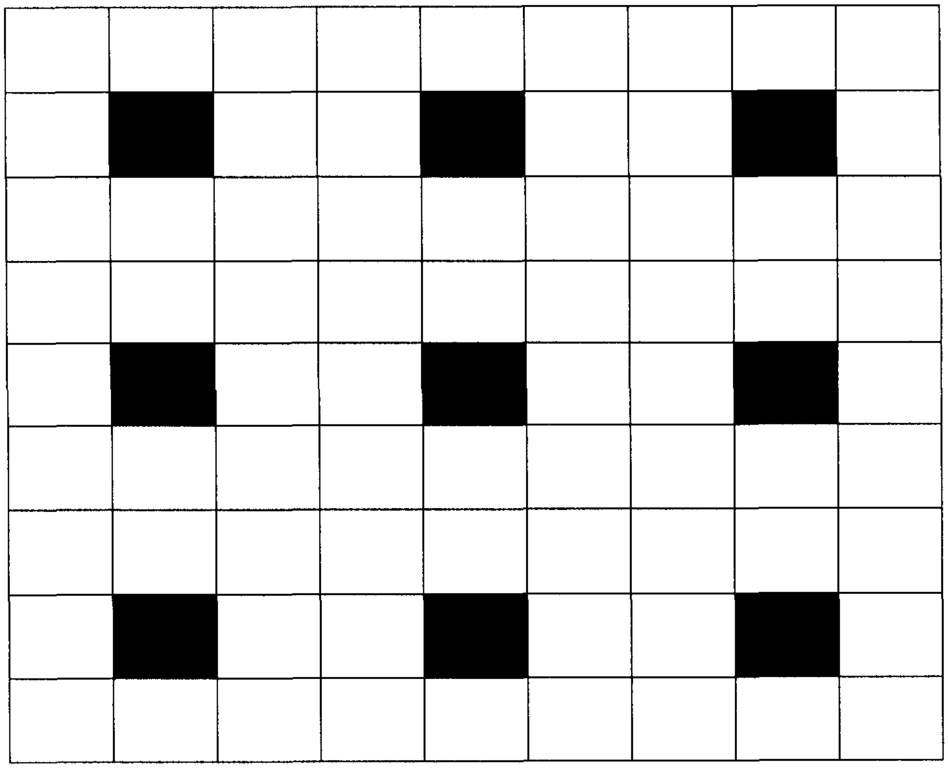

Fresh Fuel: Must be less than or equal to Nominal 4.95 w/o ^{235}U
No restrictions on burnup
Assemblies with GAD shall have a minimum of 4 fuel rods with a minimum concentration of 4.0 w/o Gd_2O_3.

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

ENCLOSURE 4

REVISION 1, EXHIBIT A FROM LAR DATED FEBRUARY 1, 2005

12 pages follow

REVISION 1
Exhibit A

LICENSEE'S EVALUATION

License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage"

1.0 DESCRIPTION

This LAR is a request to amend Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2.

The Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of the proposed Spent Fuel Pool (SFP) criticality analyses for PINGP using the Westinghouse Soluble Boron Credit Methodology. NMC also requests review and approval of the proposed changes to TS and TS Bases 3.7.17, "Spent Fuel Pool Storage", and TS 4.3, "Fuel Storage" which are supported by the proposed analyses.

2.0 PROPOSED CHANGE

This LAR proposes changes to the PINGP licensing basis by application of new SFP criticality analyses using a revised methodology.

A brief description of the associated proposed TS and TS Bases changes is provided below along with a discussion of the justification for each change. The specific wording changes to the TS and Bases are provided in Exhibits B and C.

TS Limiting Condition For Operation (LCO) 3.7.17, "Spent Fuel Pool Storage": LCO 3.7.17 defines the combination of initial enrichment, burnup and decay time for the least restrictive spent fuel storage configuration. This least restrictive configuration is referred to as the "All-Cell" configuration in the Westinghouse Electric Company, LLC (Westinghouse) analysis entitled, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality

Analysis," Reference 1. The new SFP "All-Cell" criticality analyses assume a single fuel assembly type that bounds all other fuel types. Thus, only a single figure is required in LCO 3.7.17. A new Figure 3.7.17-1 is provided for the "All-Cell" configuration based on the results of the new criticality analyses. Figure 3.7.17-2 and references to it have been deleted in the LCO statement and SR 3.7.17.1.

TS 4.3, "Fuel Storage": TS Section 4.3 provides the criteria for PINGP fuel storage including SFP criticality bases and defines more restrictive new and spent fuel storage configurations in the SFP. These more restrictive configurations are referred to as the "3x3 Array" configurations in Reference 1. References to Figure 3.7.17-2 were deleted since this figure was deleted. The new SFP "3x3 Array" criticality analyses assume two fuel assembly types: 1) fuel rods containing gadolinium (shimmed); and 2) fuel rods without gadolinium (unshimmed). These two bound all other fuel types. Thus, only two figures are required in TS 4.3.1. Figures 4.3.1-1 and 4.3.1-2 were revised to define the "3x3 Array" configuration consistent with the assumptions of the new analyses proposed in this LAR. Two new Figures 4.3.1-3 and 4.3.1-4 are provided for the "3x3 Array" configurations based on the results of the new criticality analyses. Figures 4.3.1-5 through 4.3.1-12 and references to them have been deleted. The References Section was updated to replace the SFP criticality calculation with the proposed Westinghouse analyses in Reference 1.

TS Bases 3.7.17, "Spent Fuel Pool Storage": Bases 3.7.17 have been revised to support proposed LCO 3.7.17 and incorporate the assumptions and results of Reference 1. These Bases changes are provided for information and are not part of the LAR.

In summary these changes are acceptable because they are supported by the proposed SFP criticality analyses in attached Exhibit D, Reference 1.

3.0 BACKGROUND

Spent fuel pool criticality analyses are performed to demonstrate that the spent fuel pool k_{eff} is conservatively predicted to be less than 0.95. On behalf of Westinghouse Owners Group utilities, Westinghouse developed a methodology for performing spent fuel pool criticality analyses which takes credit for soluble boron in the spent fuel pool. This methodology was documented in WCAP-14416-NP-A, Revision 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Reference 2 In 1995, Prairie Island (PI) submitted for NRC review and approval new criticality analyses to take credit for soluble boron in the PI spent fuel pool. The NRC in License Amendments 129/121 dated June 12, 1997 approved these analyses and the methodology. Although not explicitly referenced in the Prairie Island Operating Licenses or the Technical Specifications, Appendix A of the Operating Licenses, these analyses utilized the Westinghouse methodology provided in Reference 2. WCAP-14416 (Ref. 2) is referenced in the PI Updated Safety Analysis Report (USAR).

The methodology in Reference 2 utilizes a two-dimensional model of the spent fuel. To account for axial, or three-dimensional effects, a reactivity "bias" was included in the model. Another utility determined that the axial bias included in WCAP-14416 (Ref. 2) may not adequately account for the three-dimensional effects. Westinghouse performed an investigation on various aspects of the spent fuel pool criticality analyses supported by WCAP-14416 (Ref. 2). As the result of this investigation, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 00-015, Reference 3, to the affected plants. This NSAL notified the nuclear industry, including NMC, that the methodology provided in Reference 2 may be non-conservative with respect to the axial reactivity bias used to account for three-dimensional burnup effects in the two-dimensional model. The NRC also became aware of these nonconservatisms. As stated in a letter dated July 27, 2001, from the NRC to Westinghouse, the NRC staff does not view the nonconservatisms in the calculated biases as a safety concern, because of large conservatisms used in other aspects of the methodology. However, in the July 27, 2001 letter, the NRC staff also stated that

> [a]lthough this approach may lead to sufficient margin to account for the identified non-conservatism(s) on a plant specific basis, it departs from the Westinghouse methodology of WCAP-14416. Therefore, WCAP-14416 can no longer be relied upon as an approved methodology by the NRC staff or the licensees. For future licensing actions, licensees will need to submit plant-specific criticality calculations for spent fuel pool configurations that include technically supported margins.

To remove further consideration of WCAP-14416 (Ref. 2) and NSAL 00-015 (Ref. 3) for PINGP, Westinghouse performed new criticality analyses using a revised methodology, the Westinghouse Soluble Boron Credit Methodology described in Reference 1, that provides Prairie Island Nuclear Generating Plant-specific criticality calculations for spent fuel pool configurations that include technically supported margins. The results of the SFP criticality analyses support revision of LCO 3.7.17 and TS 4.3 which simplifies these Technical Specification requirements. NMC requests the NRC approve the PINGP proposed analyses, using the revised methodology, and the associated proposed TS changes. The NRC previously reviewed and approved the Westinghouse Soluble Boron Credit Methodology for other plants including the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, License Amendment Nos. 154, on September 25, 2002.

4.0 TECHNICAL ANALYSIS

PINGP is a two unit plant located on the west bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned by NSP and operated by the Nuclear Management Company (NMC). Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction

Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Prairie Island Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967.

PINGP was not licensed to NUREG-0800, "Standard Review Plan (SRP)."

Spent Fuel Pool and Stored Fuel

The spent fuel storage pool is a two compartment pool with these compartments designated as Pool 1 and Pool 2. Each pool contains spent fuel storage racks for vertical placement of new or spent fuel assemblies. 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.

The storage racks consist of storage tubes interconnected with each other through upper and lower grids which ensure the proper location of the storage tubes on 9.5 inch pitch in both directions. Each storage tube consists of three components: an inner type 304 stainless steel tube, a layer of Boraflex neutron absorbing material, and an outer skin of type 304 stainless steel. The neutron absorber material is believed to be degraded and is therefore not credited in the spent fuel pool criticality analyses.

Pools 1 and 2 are designed to accommodate new or spent fuel of various initial enrichments, burnup, decay times and numbers of gadolinium rods. Specific details of the spent fuel storage system and the fuel that are relevant to the criticality analyses are provided in Exhibit D, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis, Westinghouse Electric Company LLC, dated November 11, 2004", Reference 1. This license amendment request does not propose any physical changes to the spent fuel storage systems or other plant systems which may have an impact on storage of fuel in the SFP. Thus SFP storage events initiated external to the SFP, such as a boron dilution event, have not changed since credit for soluble boron was previously approved in License Amendment Nos. 129/121. Events initiated external to the SFP have not increased in probability, nor have different types of accidents been created, thus they are not re-evaluated in this submittal.

Licensing Basis for SFP Criticality Analyses – Acceptance Criteria

The SFP criticality analyses are required to ensure that the spent fuel pool multiplication factor, k_{eff}, is less than 0.95 as recommended by 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, Reference 4, and NRC guidance in 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, Reference 5. In addition, subcriticality of the pool (k_{eff} < 1.0) must be assured on a 95/95 (probability/confidence level) basis, without the presence of the soluble boron in the pool. NRC guidelines, based upon an accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 1.00 be evaluated in the absence of soluble boron.

The double contingency principle discussed in ANSI/ANS-8.1-1983 and the April 1978 NRC letter 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. The presence of soluble boron in the PINGP SFP is controlled by LCO 3.7.16, "Fuel Storage Pool Boron Concentration." SR 3.7.16.1 requires verification of boron concentration every 7 days which is consistent with the requirements of NUREG-1431, "Standard Technical Specifications, Westinghouse Plants."

Current Method for Criticality Analyses

The current method for PINGP SFP criticality analyses is contained in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Revision 1, November 1996 (Ref. 2). As discussed in NSAL 00-015 (Ref. 3), this methodology may be non-conservative with respect to the axial reactivity bias used to account for three-dimensional burnup effects in the two-dimensional model. Consequently, NMC in this LAR proposes new PINGP SFP criticality analyses utilizing a revised methodology.

Proposed Criticality Analyses

NMC proposes to use the analyses provided in Exhibit D, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis, Westinghouse Electric Company LLC, dated November 11, 2004", (Ref. 1) as the new SFP analyses. A brief description of the proposed analyses and the supporting revised methodology, its use and results for PINGP SFP are provided here. For a more complete description, refer to Exhibit D.

The methodology presented in Exhibit D is employed to assure the criticality safety of the SFPs and to define limits placed on fresh and depleted fuel assembly storage configurations. The analysis methodology employs SCALE-PC, a personal computer version of the SCALE-4.3 code system, and the two-dimensional integral transport code DIT (Discrete Integral Transport) with an ENDF/B-VI neutron cross section library.
The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on specific classes of personal computers. SCALE-PC includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. Benchmarking

of SCALE-PC for use in spent fuel rack criticality analyses is described in Exhibit D Section 1.3.2.

The DIT (Discrete Integral Transport) code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The multigroup cross sections utilized in DIT are based on the Evaluated Nuclear Data File Version 6 (ENDF/B-VI).

Collectively these codes demonstrate that the acceptance criteria defined in Exhibit D are met. SCALE-PC was used in benchmarking and evaluating the fuel assembly storage configurations. The DIT code is used for simulation of in-reactor fuel assembly depletion.

Basis for Proposed Licensing Basis Changes and TS Revisions

As discussed in Exhibit D, Westinghouse has modeled the PINGP spent fuel racks and their contents and performed evaluations utilizing the criticality methodology discussed above. Two fuel storage configurations, designated "All Cell" and "3x3 Array", were defined for combinations of empty storage cells, new fuel and depleted fuel with various initial enrichments, burnup, decay time and burnable poison (gadolinium) content. Fuel assemblies have been evaluated for maximum enrichments up to 5.0 weight percent (w/o).

The All Cell storage configuration is least restrictive in that empty storage cells or fuel that meets the initial enrichment, burnup and decay time requirements of proposed TS Figure 3.7.17-1 can be stored in any pattern adjacent to an empty storage cell or any other fuel assembly which meets these criteria. Based on evaluation, the Westinghouse 14x14 Standard fuel assembly was selected to be the design basis fuel assembly to represent discharged All Cell fuel assemblies.

The 3x3 Array is more restrictive in that the fuel assembly or empty location arrangement is defined in a square of three cells by three cells with a fresh assembly or an empty cell in the center storage cell as shown in proposed Figure 4.3.1-2. The fuel in the surrounding eight cells must meet the initial enrichment, burnup and decay time requirements of proposed TS Figure 4.3.1-3 or Figure 4.3.1-4. Two figures are given to account for fresh fuel assemblies with gadolinium, "shimmed", or without gadolinium, "unshimmed". Based on evaluation, the Westinghouse 14x14 Optimized fuel assembly (OFA) was selected to be the design basis fuel assembly to represent fresh fuel assemblies in the center location of the 3x3 Array and the Westinghouse 14x14 Standard fuel assembly was selected to be the design basis fuel assembly to represent peripheral discharged fuel assemblies in the 3x3 Array. An empty cell may be used in any location.

The SFP criticality acceptance criteria were met when these fuel storage configurations were evaluated applying the proposed SFP criticality methodology.

As part of demonstrating that the k_{eff} requirements are met, evaluations were performed to determine soluble boron credit requirements. A soluble boron concentration of 730 parts per million (ppm) assures that k_{eff} is less than or equal to 0.95 when accounting for burnup and reactivity depletion uncertainties and postulated accidents. For an occurrence of the postulated accident conditions, the double contingency principle discussed in ANSI/ANS-8.1-1983 and the April 1978 NRC letter (Refs. 4 and 5) can be applied. This states that the analyses are not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for the postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 730 ppm required to maintain k_{eff} less than 0.95) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event. Current SFP criticality analyses required 750 ppm to meet k_{eff} requirements (this value does not consider the additional boron required to mitigate accident induced reactivity increases). LCO 3.7.16 requires the spent fuel storage pool boron concentration to be greater than or equal to 1800 ppm whenever fuel assemblies are stored in the SFP.

Conclusions

NMC in this LAR proposes to replace the current SFP criticality methodology with the methodology presented in Exhibit D. The codes, methods and techniques contained in the methodology are used to satisfy the acceptance criteria on k_{eff}. The proposed methodology utilizes industry accepted analysis codes which have been benchmarked for SFP criticality analyses crediting soluble boron.

NMC proposes to revise LCO 3.7.17 and associated Bases and TS 4.3.1 incorporating the proposed analyses. The criticality analyses utilized two 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. These two proposed spent fuel storage configurations are defined in proposed Figures 3.7.17-1 and 4.3.1-1 through 4.3.1-4. These storage configurations correspond to the "All Cell" and "3x3 Array" configurations discussed in Exhibit D. The resulting Prairie Island spent fuel pool criticality analyses allow for the storage of fuel assemblies with enrichments up to a maximum of 5.0 weight percent U-235 while maintaining $k_{eff} \leq 0.95$ including uncertainties and credit for soluble boron.

The proposed methodology and analyses provide a conservative approach for demonstrating that the SFP will meet acceptance criteria. The proposed TS changes in conjunction with other current TS requirements assure that the spent fuel will remain subcritical during normal and postulated accident conditions. Operation of the Prairie Island Nuclear Generating Plant with these licensing basis changes and revised Technical Specifications will continue to protect the health and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

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

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   This license amendment proposes to revise the plant licensing basis by: 1) replacing the spent fuel pool criticality analyses; and 2) revising the spent fuel storage Technical Specifications 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage" utilizing the proposed analyses. The proposed Technical Specification revisions allow spent fuel to be stored in different configurations.

   The proposed changes relate to prevention of criticality accidents in the spent fuel pool. Since the current spent fuel pool criticality analyses and Technical Specifications ensure that a criticality accident does not occur, criticality accidents have not been previously evaluated. Likewise the proposed spent fuel pool criticality analyses and Technical Specifications ensure that a criticality accident does not occur. Thus the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

   Events that could cause a criticality accident were evaluated and analyses demonstrated that the current Technical Specification required soluble boron is more than adequate to assure that a criticality accident does not occur. Thus the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

   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

   This license amendment proposes to revise the plant licensing basis by: 1) replacing the spent fuel pool criticality analyses; and 2) revising the spent fuel storage Technical Specifications 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage" utilizing the proposed analyses. The proposed Technical Specification revisions allow spent fuel to be stored in different configurations.

The proposed licensing basis changes do not involve a change in system operation, or procedures involved with the fuel storage system. It does revise the allowable storage configurations. The proposed changes provide a conservative basis for evaluating spent fuel pool criticality and storage of fuel assemblies in a safe configuration which meets criticality evaluation acceptance criteria. There are no new failure modes or mechanisms created through use of the proposed analyses or proposed Technical Specifications. Use of these licensing basis changes for storage of fuel assemblies does not involve any modification in the operational limits of plant systems. There are no new accident precursors generated with use of these licensing basis changes.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

This license amendment proposes to revise the plant licensing basis by: 1) replacing the spent fuel pool criticality analyses; and 2) revising the spent fuel storage Technical Specifications 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage" utilizing the proposed analyses. The proposed Technical Specification revisions allow spent fuel to be stored in different configurations.

The proposed licensing basis change will result in a conservative calculation of the required spent fuel pool soluble boron concentration for the proposed fuel storage configurations. The current Technical Specification required spent fuel pool boron concentration significantly exceeds the proposed criticality analyses required boron concentration. The proposed analyses demonstrate that the criticality analysis acceptance criteria for the proposed fuel storage configurations are met. The proposed analyses utilize industry accepted analysis codes which have been benchmarked for the spent fuel pool criticality analyses proposed for the Prairie Island Nuclear Generating Plant. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

5.2 Applicable Regulatory Requirements/Criteria

General Design Criteria

The construction of the PINGP was significantly complete prior to issuance of 10 CFR 50, Appendix A, General Design Criteria. The PINGP was designed and constructed to comply with the Atomic Energy Commission General Design Criteria as proposed on July 10, 1967 (AEC GDC) as described in the plant Updated Safety Analysis Report (USAR). AEC GDC 66 provides design guidance for fuel storage criticality considerations.

AEC GDC proposed Criterion 66 – Prevention of Fuel Storage Criticality

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configuration shall be emphasized over procedural controls.

The spent fuel storage system is currently designed to prevent criticality through a combination of physical systems and processes. This license amendment request does not propose changes to the physical systems. This license amendment request does propose new spent fuel pool criticality analyses of the physical system and proposes new process controls for safe fuel storage configurations. The proposed analyses utilize industry accepted analysis codes which have been benchmarked for the spent fuel pool criticality analyses. The proposed analyses demonstrate that criticality is prevented by the physical storage system and the proposed fuel storage configurations.

With the changes proposed in this license amendment request, the requirements of this Criterion continue to be met.

NUREG-0800 Standard Review Plan Section 9.1.2, "Spent Fuel Storage"

The Prairie Island Nuclear Generating Plant is not licensed to the criteria listed in NUREG-0800, and nothing in the proposed amendment is intended to commit Prairie

Island Nuclear Generating Plant to the criteria in NUREG-0800.

However, Section 9.1.2 of NUREG-0800 was reviewed for guidance for evaluating the acceptability of this license amendment request. Section 9.1.2 of NUREG-0800 was written for new facilities which do not credit soluble boron. The changes proposed in this license amendment request only relate to the spent fuel pool criticality, which credits soluble boron in the storage pool, and application of the analyses to Technical Specification requirements. No physical changes are proposed with this license amendment request. Thus, NMC did not identify guidance for acceptability of this license amendment request in Section 9.1.2 of NUREG-0800.

Section 9.1.2 of NUREG-0800 applies 10 CFR 50, Appendix A, General Design Criteria (GDC) 2, 4, 5, 61, 62 and 63 as the acceptance criteria for spent fuel storage facilities. The Prairie Island Nuclear Generating Plant construction was significantly complete prior to issuance of these criteria and thus is not committed to meet them. As discussed above, the Prairie Island Nuclear Generating Plant meets AEC GDC proposed Criterion 66.

Regulatory Requirements/Criteria 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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Westinghouse Electric Company, LLC calculation CN-WFE-03-40, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis", dated November 11, 2004".

2. WCAP-14416-NP-A, Revision 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology."

3. Nuclear Safety Advisory Letter (NSAL) 00-015, November 2, 2000.

4. 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.

5. Nuclear Regulatory Commission Letter from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.