February 26, 2002

Mr. Douglas E. Cooper
Site Vice President
Palisades Plant
Nuclear Management Company, LLC
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES PLANT - ISSUANCE OF AMENDMENT TO CHANGE ENRICHMENT LIMITS IN THE FUEL POOL (TAC NO. MB1362)

Dear Mr. Cooper:

The Commission has issued the enclosed Amendment No. 207 to Facility Operating License No. DPR-20 for the Palisades Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to the application by Consumers Energy Company (CEC) dated March 2, 2001, as supplemented March 29, September 14, and December 27, 2001. CEC has subsequently been succeeded by Nuclear Management Company, LLC (NMC), as the licensed operator of the Palisades Plant. By letter dated May 17, 2001, NMC requested that the Commission continue to process and disposition licensing actions previously docketed and requested by CEC.

The amendment changes the TSs to increase the limits on stored fuel enrichments and provide other more flexible fuel loading constraints for the storage racks for new and spent fuel.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/ by B. Mozafari

Darl S. Hood, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures: 1. Amendment No. 207 to DPR-20
2. Safety Evaluation

cc w/encls: See next page

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<u>DISTRIBUTION</u>

PUBLIC OGC GHill(2)
PDIII-1 Reading ACRS AVegel, RGN-III
WReckley RDenning
DHood YOrechwa
RBouling GHatchett

*Provided SE input by memo
**Previously Concurred

OFFICE	PDIII-1/PM	PDIII-1/LA	SPLB/(A)SC*	SRXB/SC*	RTSB/SC**	OGC	PDIII-1/SC(A)
NAME	DHood	RBouling	BThomas	FAkstulewicz	RDennig	AHodgdon	WReckley
DATE	02/22/02	02/22/02	01/14/02	12/10/01	02/18/02	02/22/02	02/26/02

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. DPR-20

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated March 2, 2001, as supplemented March 29, September 14, and December 27, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-20 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 207, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William D. Reckley, Acting Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 26, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A of the Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.7.15-1	3.7.15-1
3.7.16-1	3.7.16-1
3.7.16-2	3.7.16-2
4.0-1	4.0-1
4.0-2	4.0-2
4.0-3	4.0-3
---	4.0-4
B 3.7.15-1	B 3.7.15-1
B 3.7.15-2	B 3.7.15-2
B 3.7.16-1	B 3.7.16-1
B 3.7.16-2	B 3.7.16-2
B 3.7.16-3	B 3.7.16-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-20

NUCLEAR MANAGEMENT COMPANY, LLC

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated March 2, 2001, as supplemented March 29, September 14, and December 27, 2001, the licensee[1] requested an amendment to change the Technical Specifications (TSs) for the Palisades Plant. The proposed amendment would change the limiting conditions for operation (LCOs), surveillance requirements (SRs), and design features in the TSs to provide more flexible fuel loading constraints for the Palisades fuel storage racks and accommodate future core designs. The changes would affect TS Sections 3.7.15, "Spent Fuel Pool (SFP) Boron Concentration," 3.7.16, "Spent Fuel Assembly Storage," and 4.3, "Design Features--Fuel Storage." Allowed uranium enrichments for storage would be increased.

Specifically, enrichment limits for fuel racks for the storage of new, unirradiated fuel[2] (currently limited to fuel assemblies having a maximum average planar uranium-235 (U-235) enrichment of 4.20 weight-percent) would be increased to allow storage of 24 unirradiated fuel assemblies having a maximum planar average U-235 enrichment of 4.95 weight-percent, subject to proposed loading pattern constraints (e.g., the center row being empty if stored fuel exceeds 4.05 weight-percent U-235 enrichments). Similarly, the new fuel storage racks could contain 36 unirradiated fuel assemblies having a maximum planar average U-235 enrichment of 4.05 weight-percent, subject to similar proposed loading pattern constraints not necessarily requiring the center row to be empty. Region I fuel storage racks (currently limited to a maximum enrichment of 4.40 weight-percent) would be changed to allow storage of unirradiated or irradiated fuel up to 4.95 weight-percent enrichment on the basis of revised criticality analyses that assume no credit for soluble boron in the pool under normal conditions, but which take credit for 1350 parts per million (ppm) of soluble boron under accident conditions. Enrichment requirements for Region II fuel storage racks (currently limited to 3.27 weight-percent) would be changed to allow storage of unirradiated fuel up to 1.14 weight-percent and irradiated fuel of equivalent reactivity up to 4.6 weight-percent initial

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[1]The March 2, 2001, application and March 29, 2001, supplemental letter were submitted by the Consumers Energy Company (CEC). CEC has subsequently been succeeded by the Nuclear Management Company, LLC (NMC), as the licensed operator of Palisades. By letter dated May 17, 2001, NMC requested that the Commission continue to process and disposition licensing actions previously docketed and requested by CEC.

[2] The existing fuel racks installed at the facility for the storage of new, unirradiated fuel are hereafter referred to as new fuel storage racks. This term does not refer to new racks.

enrichment on the basis of criticality analyses that take credit for 850 ppm of soluble boron in the pool under normal conditions and 1350 ppm of soluble boron under accident conditions. The TSs (e.g., proposed Table 3.7.16-1) for allowable enrichments for fuel storage in Region II of the spent fuel pool (SFP) or the north tilt pit would continue to be based upon a combination of initial enrichment and burnup, but the proposed change would also add decay time to this combination. The existing limitations that Region I spent fuel racks may contain only “new or partially spent” fuel assemblies, and that Region II spent fuel racks may contain only “partially spent” fuel assemblies would be changed to “new or irradiated fuel assemblies which meet the initial enrichment, burnup, and decay time requirements of [the proposed revision to] Table 3.7.16-1.” The existing requirements that fuel assemblies in new or Region I fuel storage racks must contain “216 rods which are either UO_2, $Gd_2O_3UO_2$, or solid metal” would be deleted. TS 3.7.15 would continue to require that the SFP boron concentration be equal to or greater than 1720 ppm whenever fuel is stored in the SFP, and be verified weekly. However, the optional Action Statement A.2.2 to immediately initiate action to perform an SFP verification when the concentration is not within limits would be deleted (as would a related portion of the applicability statement regarding verification). The licensee also included changes to the associated TS Bases.

2.0 BACKGROUND

In Appendix A to Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), General Design Criterion (GDC) 62, states that “Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.” The Nuclear Regulatory Commission (NRC) has established a 5-percent subcriticality margin (i.e., k_{eff} is to be no greater than 0.95) to comply with GDC 62.

10 CFR 50.68, “Criticality accident requirements,” states that “If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.”

On June 26, 1996, the NRC staff issued Generic Letter (GL) 96-04, “Boraflex Degradation in Spent Fuel Pool Storage Racks,” addressing concerns related to (1) gamma radiation-induced shrinkage of Boraflex and the potential to develop tears or gaps in the material, and (2) long-term Boraflex performance throughout the intended service life of the racks resulting from gamma irradiation and exposure to the wet pool environment. In GL 96-04, the NRC staff requested licensees that use Boraflex to assess the ability of Boraflex to maintain a k_{eff} of 0.95, and to submit a plan describing actions required if the 5-percent margin to criticality could not be maintained by Boraflex material due to current or projected material degradation.

In a safety evaluation dated October 25, 1996, the NRC staff accepted Westinghouse Owners Group Topical Report WCAP-14416-A, “Westinghouse Spent Fuel Rack Criticality Analysis Methodology,” for the purpose of referencing the report in licensing applications where licensees propose to take credit for soluble boron in SFP criticality analyses. The NRC staff’s review and acceptance of WCAP-14416-A focused on the methodology whereby credit could be taken for soluble boron in the SFP to meet the NRC’s recommended criterion that the spent fuel rack k_{eff} be less than or equal to 0.95, at a 95-percent probability, 95-percent confidence level. All licensees proposing to use this method for soluble boron credit were requested to

(1) identify potential events that could dilute the SFP soluble boron to the concentration required to maintain the k_{eff} limit of 0.95, and (2) quantify the time span of these dilution events to show that sufficient time would be available to enable adequate detection and suppression of any dilution event.

Accordingly, the licensee is proposing to modify Palisades TSs 3.7.15, 3.7.16, and 4.3 to increase fuel enrichment limits and take credit for soluble boron in the spent fuel criticality analysis to allow greater flexibility in fuel storage. The NRC staff has evaluated the proposed changes with respect to potential effects upon criticality analyses, boron dilution events, and pool cooling capacity. The NRC staff's evaluation is presented in Section 3.0 below.

3.0 EVALUATION

3.1 Criticality Analyses

For the purpose of evaluating the licensee's amendment request, the proposed changes can be divided into three general areas:

1. Allowing storage of un-irradiated fuel with enrichment of up to 4.95 weight-percent U-235 in new fuel storage racks according to predefined loading patterns.

2. Allowing storage of un-irradiated or irradiated fuel with enrichment of up to 4.95 weight-percent U-235 in Region I fuel storage racks with no credit for soluble boron in the pool under normal conditions, and credit for 1350 ppm of soluble boron under accident conditions.

3. Allowing storage of un-irradiated fuel with enrichment of up to 1.14 weight-percent U-235 and irradiated fuel of equivalent reactivity up to 4.6 weight-percent U-235 initial enrichment in Region II fuel storage racks with credit for 850 ppm of soluble boron in the pool under normal conditions, and credit for an additional 500 ppm of soluble boron (total of 1350 ppm) under accident conditions. Assembly burnup and subsequent decay time are considered in the criticality calculations of Region II. The Region II fuel storage rack criticality analyses conservatively ignore the Boraflex poison material present in the racks.

The criticality analyses justifying these changes closely follow the NRC-approved methodology described in WCAP-14416-NP-A. The criticality analyses performed by the licensee, however, differ in that the Monte Carlo code package, MONK7A, is employed as opposed to KENO-Va, and the depletion code, CASMO-3, as opposed to PHOENIX. These alternate computer codes, when adequately benchmarked, are acceptable within the NRC guidance on the regulatory requirements for criticality analysis of fuel storage at light-water reactor power plants (i.e., see memorandum to T. Collins, NRC, from L. Kopp, NRC, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998).

3.1.1 Analytic Methods

The analysis methods used by the licensee to assess criticality in the Palisades fuel pool differ from those presented in the NRC-approved Westinghouse methodology described in WCAP-14416-NP-A. The licensee used the Monte Carlo computer code, MONK7A, rather than KENO-Va for computing the value of k_{eff} in the fuel pool, and CASMO-3 rather than PHOENIX for establishing reactivity equivalence between a fuel assembly burnup increment and an equivalent allowable enrichment increment. However, subsequent to NRC's approval of WCAP-14416-NP-A, NRC staff guidance (see NRC guidance by memorandum dated August 19, 1998) has included the codes MONK6B (together with the 8220-group United Kingdom Nuclear Data Library (UKNDL)), and the CASMO-3 code, in the set of acceptable computer codes for performing criticality analysis of fuel storage at light-water reactor power plants. To assure that MONK7A, an update of MONK6B, conserves all of the major features of MONK6B, extensive validation of MONK7A has been performed by the vendor of the MONK code system (see "MONK: A Monte Carlo Program for Nuclear Criticality Safety Analysis - User's Guide for Version 7A," Issue 3, AEA Technology, July 1996). As was the case with MONK6B, the code MONK7A is shown to systematically over-predict k_{eff} for enriched UO_2 systems when using the UKNDL cross-section library. The extensive validation shows that the over-prediction ranges from 0.01 Δk to 0.0003 Δk. In particular, the licensee has also compared MONK6B to MONK7A k_{eff} computations for four key sets of critical experiments relevant to low-enriched uranium fuel storage, such as that found at Palisades. Over the range of parameters of interest to this analysis, the licensee has presented the results of 10 additional comparison calculations of k_{eff} between MONK7A and critical experiments. These latter two comparisons validate the licensee's analytic tools, while the former--those performed by the vendor of the code--demonstrate statistically consistent results which assure a systematic over-prediction by MONK7A of k_{eff} and give a conservative bound. Given the extent of the validation of MONK7A, the NRC staff finds the use of the MONK7A computed value of k_{eff} to be a conservative estimate of k_{eff} and that its use is acceptable.

The relative changes in k_{eff}, Δk (due either to the variation in manufacturing dimensions or from fuel assembly burnup) are computed with the two-dimensional, multi-group transport theory code, CASMO-3 (see "CASMO-3: A Fuel Assembly Burnup Program; User's Manual," Studsvik, November 1, 1994). The licensee has NRC approval for the use of COSMO-3 for the calculation of fuel cross sections for the core monitoring software PIDAL (see "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Revision of the PIDAL In-Core Monitoring Code," Docket No. 50-255, May 6, 1997); and also has extensive experience in the application of CASMO-3 in k_{eff} calculations for the evaluation of control rod worths, estimation of boron requirements and core design. In addition, the licensee validated the CASMO-3 model by comparing the values of k_{∞} calculated for Region I and Region II of the Palisades fuel pool via COSMO-3, MONK7A, and KENOVa. In lieu of an analysis to estimate the uncertainty in the reactivity associated with fuel depletion, the licensee has conservatively taken the uncertainty as 5 percent of the reactivity decrement associated with the burnup. To account for the reactivity effects from the axial burnup profile in the two-dimensional CASMO-3 calculations, the licensee has evaluated the axial burnup profiles of discharged assemblies in the Palisades fuel pool and developed a set of adjustment factors that are used to bias the k_{eff} in the burnup equivalence calculations. The NRC staff concludes that these considerations are consistent with the aforementioned NRC guidelines by memorandum dated August 19, 1998, and are, therefore, acceptable.

3.1.2 New Fuel Storage Racks

The rack for storage of new fuel was designed for storage of 72 assemblies under the assumption of normal dry conditions and without an allowance in spacing or poisoning to take into account the increase in reactivity due to the addition of a moderator. The licensee intends to meet the regulatory requirement of SFP subcriticality (k_{eff} < 0.95) at optimum moderation conditions for two new loading patterns through a combination of restrictions on the maximum planar average fissile enrichment of the fuel, the number of fueled assemblies, and the loading pattern of these assemblies in the rack. To this end, the licensee presents criticality analyses for two variants for fuel assembly storage in the new fuel storage racks:

a. 24 unirradiated fuel assemblies with a maximum planar average U-235 enrichment of 4.95 weight-percent, and stored in accordance with the pattern shown in Figure 4.3 -1 of the proposed TS.

b. 36 unirradiated fuel assemblies with a maximum planar average U-235 enrichment of 4.05 weight-percent, and stored in accordance with the pattern shown in Figure 4.3 -1 of the proposed TS.

The licensee computes k_{eff} at a 95-percent probability and a 95-percent confidence level (95/95 k_{eff}) as the sum of two components: the MONK7A computed k_{eff} at the worst case manufacturing tolerances and water at optimum moderator density; and two times the standard deviation of the Monte Carlo error in the MONK7A computed k_{eff}. This approach is consistent with NRC guidance by memorandum dated August 19, 1998, and is, therefore, acceptable. The 95/95 k_{eff} values for the above two variants of fuel assembly storage in the new fuel storage racks are 0.9361 for the first and 0.9482 for the second. The NRC staff concludes that both values conform to the requirements of 10 CFR 50.68, "Criticality accident requirements," and are, therefore, acceptable as a basis for the proposed changes to TS 4.3.1.3.

3.1.3 Region I Fuel Storage Racks

The Region I spent fuel storage rack design was previously qualified for storage of 15X15 fuel assemblies with maximum enrichments up to 4.40 weight-percent U-235 (see Siemens Nuclear Power Corporation Report EMF-91-174(P), "Criticality Safety Analysis for the Palisades Spent Fuel Storage Pool NUS Racks," October 1991). The Region I racks have two separate geometries, with the "main pool" rack geometry being more limiting. Thus, it is sufficient to use this more limiting geometry in the criticality evaluation of the Region I racks. Some assumptions specific to the Region I criticality calculations are: All storage cells contain fresh fuel assemblies with an enrichment of 4.95 weight-percent U-235; the SFP moderator is water without soluble boron and a water density of 1.0 gm/cm^3; the Boron-10 (^{10}B) loading in the rack poison sheets is conservatively modeled as less than the minimum manufacturers reported areal density of 0.0959 g/cc; a reactivity bias of 0.0012 is applied to account for the effect of the normal range of fuel pool water temperatures of 40 °F to 150 °F.

The computed 95/95 k_{eff} of the Region I racks under normal conditions is 0.9449. This value takes into account all mechanical and computational uncertainties and assumes 0.0 ppm of natural boron in the fuel pool storage water. The NRC staff concludes that the value conforms to the requirements of 10 CFR 50.68 and is, therefore, acceptable as a basis for the proposed changes to TS 4.3.1.1.

3.1.4 Region II Fuel Storage Racks

The Region II spent fuel storage racks were previously qualified for storage of 15X15 fuel assemblies with enrichments up to 3.27 weight-percent U-235 and conforming to the applicable initial enrichment vs. burnup requirements (see Westinghouse Report WNEP-8626, Revision 2, "Design Report of Region Two Spent Fuel Storage Racks: Plant Applicability - Consumers Power Company Palisades Plant," Westinghouse Electric Corporation Nuclear Components Division, Pensacola, FL, May 1987). A unique characteristic of the Region II racks at Palisades is the use of the neutron absorber, Boraflex, as an integral part of the rack. In the criticality analysis of the Region II racks, no credit is taken for the presence of Boraflex. The consequent loss of some negative reactivity is compensated in the calculations by taking credit for the presence of soluble boron in the fuel pool water, and increasing the required assembly burnup for a given enrichment. Some of the conservative assumptions introduced in the k_{eff} computations of the Region II racks are: The reactivity calculation for spent fuel ignores Xe; the Boraflex volume is replaced with pure water; conservative reactivity adjustments based on limiting axial burnup profiles are assumed for each burnup/enrichment point on the burnup credit curve; conservative biases are introduced to account for uncertainty in computational methodology, water temperature, soluble boron credit, plant exposure records, and the reactivity equivalence methodology.

The Region II 95/95 k_{eff} is computed by adding the temperature bias and the statistical sum of independent tolerance uncertainties to the nominal (without credit for soluble boron) MONK reference reactivity. The computed value is 0.9987, which is consistent with the requirement in 10 CFR 50.68 that the fuel racks remain subcritical ($k_{eff} < 1.0$) when no soluble boron is present in the SFP water.

The requirement of 10 CFR 50.68 that the 95/95 k_{eff} be less than 0.95 is met by taking credit for 850 ppm of soluble boron in the SFP. This computed 95/95 k_{eff}, which takes additional account of biases related to reactivity equivalencing, boron credit methods, and uncertainty in plant exposure records, is 0.9478.

The NRC staff concludes that the two computed 95/95 k_{eff} values conform to the requirements of 10 CFR 50.68 and are, therefore, acceptable as a basis for the proposed changes to TS 4.3.1.2.

In its supplemental letter dated September 14, 2001, the licensee noted that programs designed to ensure the integrity of the Boraflex poison in the Region II fuel storage racks (e.g., periodic blackness testing) will be discontinued. Since the criticality analyses submitted in support of this amendment do not credit the Boraflex poison material in the Region II racks, and hence the Boraflex is no longer relied upon for reactivity control, the NRC staff agrees that periodic blackness testing of the Boraflex need not be continued at Palisades.

3.1.5 Fuel Elevator/Fuel Transfer Machine

Both the Palisades fuel elevator and the fuel transfer machine were previously qualified for holding two 15X15 fuel assemblies with a maximum planar average enrichment of up to 4.40 weight-percent U-235 and a minimum pool boron concentration of 600 ppm. The licensee has presented criticality calculations for two fuel assemblies in the transfer machine, each with a 4.95 weight-percent U-235 enrichment, which show that 850 ppm of boron in the pool water ensures that the 95/95 k_{eff} remains below 0.95. Similarly, the licensee has shown that for normal operations in the elevator/inspection station, no boron is required to maintain a 95/95 k_{eff} below 0.95. The licensee has also shown that in an unlikely close approach of an assembly in a raised elevator, 850 ppm of boron in the pool water is sufficient to maintain a 95/95 k_{eff} below 0.95. Thus, the proposed TS changes are acceptable with respect to the fuel elevator and the fuel transfer machine.

3.1.6 Rack Interaction, Abnormal Conditions, and Postulated Accidents

The licensee's criticality calculations for the interaction between storage racks and/or fuel handling equipment shows that a fuel pool boron concentration above 850 ppm is sufficient to insure the required 95/95 k_{eff} is below 0.95. The limiting accident condition is the misloading of a fuel assembly. In this event, an additional 500 ppm of soluble boron will mitigate the effect of the event. Thus, the overall requirement for the fuel pool boron concentration to ensure that the 95/95 k_{eff} remains below 0.95 under all normal and credible accident scenarios is 1350 ppm. The 1350 ppm accident requirement is 500 ppm above the 850 ppm concentration required in the proposed TS 4.3.1.2.c, and is conservative with respect to the TS of a minimum fuel pool boron concentration of 1720 ppm. The use of the double contingency principle (discussed in the aforementioned NRC guidance memorandum dated August 19, 1998), along with the requirements imposed by LCO 3.9.1, LCO 3.7.15, SR 3.7.15.1, and SR 3.9.1.1, will ensure that adequate soluble boron will be maintained in the fuel pool water at all times. Therefore, the NRC staff concludes the proposed TS changes are acceptable with respect to potential interactions between the storage racks and/or fuel handling equipment, including misloading events.

3.2 Boron Dilution Analysis

The licensee has proposed to credit soluble boron in the criticality analysis which will allow the neutron absorbing properties of the Boraflex material in the fuel storage racks to be ignored. Therefore, the licensee completed a boron dilution analysis to support crediting soluble boron and its proposed revision to TS Section 3.7.15. As a result, the licensee established a boron concentration of greater than or equal to 1720 parts per million for fuel assembly storage and movement within the fuel storage pool. In order to ensure that the design-basis k_{eff} of 0.95 is not exceeded due to potential dilution events, the licensee determined that a minimum boron concentration of 850 ppm would provide a k_{eff} of less than or equal to 0.95. The licensee then evaluated plant systems that could dilute the SFP.

The Palisades SFP has a water inventory of 165,300 gallons. The licensee stated that the volume required to dilute the SFP is 123,007 gallons, which corresponds to the minimum SFP soluble boron concentration of 850 ppm. The volume required to dilute the SFP soluble boron concentration is based upon the initial concentration of 1720 ppm.

The licensee evaluated 11 dilution scenarios of which 7 are considered Category 1 events and four are Category 2 events. The licensee's submittal did not adequately address the guidelines of WCAP-14416-A for crediting soluble boron in the SFP, as accepted by the NRC staff in its evaluation dated October 25, 1996. Specifically, the submittal did not provide specific information regarding dilution sources, dilution flow rates, boration sources, administrative procedures, instrumentation, boron dilution initiating events, and boron dilution times and volumes.

By supplemental letter dated December 27, 2001, the licensee responded to an NRC staff request for additional information by providing information similar to the guidelines of WCAP-14416-A. The licensee's response also provided a description of Category 1 and 2 events.

Category 1 events are considered credible dilution events and Category 2 events are considered incredible events. Category 2 events are deliberate dilution events in response to a loss of shielding (water) from the SFP in which the operators manually align systems to restore SFP level. To be conservative in the dilution analysis, the licensee assumed that while the SFP level is being restored, the operators allow the SFP to overflow in response to a Category 2 event.

Based on the Category 1 events evaluated, there was only one system with practically an infinite water storage source (Lake Michigan) that could provide the 123,007 gallons of water needed to dilute the SFP. Lake Michigan supplies the 1½ inch fire hose station with the 123,007 gallons needed to dilute the SFP soluble boron concentration. However, at a flow rate of 210 gpm, it would take 9.8 hours to dilute the SFP soluble boron concentration from 1720 ppm to 850 ppm.

The NRC staff verified the licensee's calculation for the addition of fire water via the 1-½ inch hose station and determined that it would take 116,511 gallons and 9.26 hours to dilute the pool soluble boron concentration from 1720 ppm to 850 ppm. However, the licensee's operating practice requires at least one operator round each 8-hour shift. Thus, if an SFP dilution were to occur from this system, reasonable assurance exists that it would be identified and suppressed by the operator before the 0.95 k_{eff} limit is reached. As an additional measure, the licensee has committed to add a fuel pool high level alarm to give an earlier warning of fuel pool increases which could lead to dilution of the soluble boron concentration.

The licensee concluded that an unplanned or inadvertent event that would dilute the SFP is not credible for Palisades. The NRC staff finds that the combination of the large volume of water required for a dilution event, the operating practice of an operator round every 8 hours, flow rates and dilution times, the licensee's administrative requirements, and TS-controlled SFP concentration and 7-day sampling requirement are adequate to detect a dilution event prior to k_{eff} reaching 0.95. Therefore, the analysis, administrative controls, and proposed TS requirement in TS 3.7.15 are acceptable for ensuring that sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum TS boron concentration to the boron concentration required to maintain the 0.95 k_{eff} design-basis limit.

In TS 3.7.15, the licensee proposes to change the existing Applicability and Required Action Statements. Specifically, the Applicability statement would be changed from "When fuel assemblies are stored in the SFP and a verification of the stored assemblies has not been performed," to read "When fuel assemblies are stored in the Spent Fuel Pool." This is a more restrictive change because the revised TS would apply anytime fuel is in the SFP. The current Action required by TS 3.7.15 if the SFP boron concentration is not within the specified limit is to immediately suspend movement of fuel assemblies in the SFP and either (1) immediately initiate action to restore the SFP boron concentration to within the specified limit, or (2) immediately initiate action to perform an SFP verification. Because the Applicability statement of TS 3.7.15 would be changed by deleting the condition that "a verification of the stored assemblies has not been performed," the existing alternative action to immediately initiate action to perform an SFP verification would no longer apply and should be deleted. Moreover, the NRC staff recognizes that, because some credit for dissolved boron in the SFP is assumed in determining the margins available in the criticality analyses for the proposed amendment, the existing alternative action to immediately verify that fuel is in the proper locations (in lieu of immediately restoring boron concentration) may not adequately ensure against required margins to criticality in the event of significant dilution of the SFP. Therefore, the NRC staff finds the proposed changes to TS 3.7.15 to be appropriate and acceptable.

3.3 Pool Cooling Effects

The NRC staff evaluated the effects of the proposed fuel enrichment to 4.95 weight-percent on the decay heat removal capacity of the SFP cooling system. The SFP cooling system is designed to maintain water clarity and remove decay heat from the SFP. As noted by the licensee in its supplemental letter dated December 27, 2002, the determination of the decay heat load is primarily a function of the operational power and burnup and is not affected by the initial fuel enrichment. As such, higher enrichments might allow a longer operating cycle that could impact the SFP cooling systems resulting from the increased decay heat. However, the licensee is not proposing to extend the Palisades operating cycle. Consequently, the decay heat load on the SFP cooling system is essentially unchanged. Therefore, the proposed increase in fuel enrichment will have an insignificant or no impact upon the SFP cooling system's ability to meet its intended design function.

3.4 Summary of Conclusions

On the basis of its review as discussed in Section 3.1 above, the NRC staff finds that the proposed changes to the Palisades TSs regarding fuel storage are based upon acceptable criticality analyses and the proposed changes are, therefore, acceptable with respect to criticality. On the basis of its review as discussed in Sections 3.2 and 3.3 above, the NRC staff's independent calculation of boron dilution times and volumes, review of the licensee's evaluation, and the NRC staff's experience gained from review of potential dilution of SFP soluble boron concentration and fuel enrichment in similar applications for other nuclear power plants, the NRC staff finds the licensee's boron dilution analysis, and the effects of fuel enrichment on the SFP heat generation, to be acceptable. The proposed TS changes are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official agreed with the NRC staff's proposed issuance of the amendment.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 10 CFR 51.32, and 10 CFR 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on December 19, 2001 (66 FR 65516), for this amendment. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based upon the considerations discussed above, that: (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.

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