April 1, 2003

Mr. J. A. Price
Site Vice President - Millstone
Dominion Nuclear Connecticut, Inc.
c/o Mr. David W. Dodson
Rope Ferry Road
Waterford, CT 06385

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: SPENT FUEL POOL REQUIREMENTS (TAC NO. MB3386)

Dear Mr. Price:

The Commission has issued the enclosed Amendment No. 274 to Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2, in response to your application dated November 6, 2001, as supplemented on December 27, 2001, and July 15, August 6, and October 29, 2002.

The amendment revises the Technical Specifications associated with the spent fuel pool (SFP). Specifically, the amendment increases the allowable nominal average fuel assembly enrichment from 4.5 weight percent (w/o) Uranium-235 (U-235) to 4.85 w/o U-235 for all regions of the SFP, the new fuel storage racks (dry), and the reactor core; allows fuel to be located in the 40 storage cells in Region B of the SFP that are currently empty and blocked; credits SFP soluble boron for reactivity control during normal conditions; and reduces the Boraflex reactivity credit in Regions A and B of the SFP.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 274 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

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Unit 2

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DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 274
License No. DPR-65

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

   A. The application for amendment by the applicant dated November 6, 2001, as supplemented on December 27, 2001, and July 15, August 6, and October 29, 2002, 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; and

   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 this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 274, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 1, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 274

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
IX	IX
XIV	XIV
3/4 9-20	3/4 9-20
3/4 9-21	3/4 9-21
3/4 9-22	3/4 9-22
3/4 9-23	3/4 9-23
3/4 9-23a	3/4 9-23a
3/4 9-24	3/4 9-24
3/4 9-25a	3/4 9-25a
3/4 9-26	3/4 9-26
B 3/4 9-3b	B 3/4 9-3b
B 3/4 9-4	B 3/4 9-4
5-4	5-4
5-5	5-5
5-5a	5-5a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 274

TO FACILITY OPERATING LICENSE NO. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By application dated November 6, 2001, as supplemented on December 27, 2001, and July 15, August 6, and October 29, 2002, Dominion Nuclear Connecticut, Inc., (the licensee), requested a change to the Millstone Power Station, Unit No. 2 (MP2) Technical Specifications (TSs).

The supplement dated December 27, 2001, provided a revision to the licensee's analysis of the issue of no significant hazards consideration, as originally provided in the November 6, 2001, application. The supplements dated July 15, August 6, and October 29, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 19, 2002 (67 FR 7414).

The proposed amendment would revise the TSs associated with the spent fuel pool (SFP) in order to:

1) increase the allowable nominal average fuel assembly enrichment from 4.5 weight percent (w/o) Uranium-235 (U-235) to 4.85 w/o U-235 for all regions of the SFP, the new fuel storage racks (dry), and the reactor core;

2) allow fuel to be located in the 40 storage cells in Region B of the SFP that are currently empty and blocked;

3) credit SFP soluble boron for reactivity control during normal conditions; and

4) reduce the Boraflex reactivity credit in Regions A and B of the SFP.

The licensee's application provided the following reasons for the above proposed changes:

Increasing the allowable nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 will allow for more flexibility in cycle length designs. Also, it will potentially reduce the amount of spent fuel generated by allowing the potential for reduced feed fuel batch sizes.

At present, MP2 is in cycle 14 operation. The MP2 SFP will lose the capacity for full core offload of the reactor core at the end of cycle 15. The proposed change recovers the use of the 40 currently blocked cells for the storage of spent fuel. If fuel can be located in the 40 cells currently blocked in Region B of the SFP, the loss of full core reserve will be delayed until the end of cycle 16.

Credit for soluble boron for reactivity control under normal conditions is primarily needed to offset the reduced Boraflex reactivity credit. In addition, reduced Boraflex reactivity credit in Region A and B of the SFP would be implemented to make allowance for the possibility of future degradation of the Boraflex. Reducing the amount of Boraflex reactivity credit taken would allow time to respond should Boraflex degradation be detected by the inservice testing (IST) program.

The specific proposed TS changes are as follows:

1.1 TS 3.9.16.2, "Shielded Cask"

This TS currently requires that the SFP boron concentration be greater than or equal to 800 parts per million (ppm) prior to movement of a shielded cask over the SFP cask laydown area. The licensee proposes to delete TS 3.9.16.2 and merge the SFP boron concentration requirements for movement of shielded casks with the SFP boron concentration requirements associated with fuel assemblies and consolidated fuel storage boxes (CFSBs) as part of the proposed changes to TS 3.9.17 as discussed in Safety Evaluation (SE) Section 1.2.

1.2 TS 3.9.17, "Movement of Fuel in Spent Fuel Pool"

TS 3.9.17 currently is titled "Movement of Fuel in Spent Fuel Pool." The title would be changed to "Spent Fuel Pool Boron Concentration."

The current Limiting Condition for Operation (LCO) requires that the SFP boron concentration be greater than or equal to 800 ppm whenever a fuel assembly or a CFSB is <u>moved</u> in the SFP. The proposed LCO would require that the SFP boron concentration be greater than or equal to 1720 ppm whenever a fuel assembly or a CFSB is <u>stored</u> in the SFP.

The current Action Statement requires that all fuel movement be suspended with the boron concentration less than 800 ppm. The proposed Action Statement would require the suspension of movement of all fuel, CFSBs, and shielded casks with the boron concentration less than 1720 ppm, and the immediate initiation of action to restore the SFP boron concentration to within its limit. The Action Statement would also be revised to state that the provisions of TS 3.0.3 are not applicable.

The current Surveillance Requirement (SR) requires that the boron concentration be verified to be greater than or equal to 800 ppm within 24 hours prior to any movement of a fuel assembly or a CFSB in the SFP, and every 72 hours thereafter. The proposed SR would require that the boron concentration be verified to be greater than or equal to 1720 ppm every 7 days, and within 24 hours prior to any movement of a fuel assembly or a CFSB in the SFP, or shielded cask over the cask laydown area.

1.3 TS 3.9.18, “Spent Fuel Pool - Reactivity Condition”

TS 3.9.18 is currently titled “Spent Fuel Pool - Reactivity Condition.” The title would be changed to “Spent Fuel Pool - Storage.”

The licensee proposes extensive changes to TS 3.9.18 in order to make it consistent with the improved Standard Technical Specifications (STS), NUREG-1432, Revision 2, “Standard Technical Specifications, Combustion Engineering Plants” TS 3.7.18. The licensee's submittal states that the proposed changes are meant to be TS improvements, are intended for clarification purposes, and that none of the proposed changes were necessary to support the underlying design changes associated by this license amendment request.

1.4 Figure 3.9-1A, “Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C”

This TS figure shows the minimum required fuel assembly exposure as a function of initial enrichment to permit storage of fuel assemblies in SFP Region C. This TS figure would be revised to reflect the revised criticality analysis.

1.5 Figure 3.9-1B, “Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C with Poison Pins Installed”

This TS figure shows the minimum required fuel assembly exposure as a function of initial enrichment to permit storage of fuel assemblies in SFP Region C, with poison pins installed. This TS figure would be revised to reflect the revised criticality analysis.

1.6 Figure 3.9-2, “Spent Fuel Pool Arrangement”

This TS figure shows the arrangement of the SFP. This TS figure would be revised to more clearly show the SFP arrangement. There are no technical changes proposed to the figure.

1.7 Figure 3.9-4, “Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region A”

This TS figure shows the minimum required fuel assembly exposure as a function of initial enrichment to permit storage of fuel assemblies in SFP Region A. This TS figure would be revised to reflect the revised criticality analysis.

1.8 TS 3.9.19, “Spent Fuel Pool - Storage Pattern”

TS 3.9.19 provides the fuel storage pattern limitations for SFP Region B. The LCO would be revised to reflect that the blocked locations in Region B may store a Batch B fuel assembly underneath the cell blocker. A footnote would be added to clarify that a Batch B fuel assembly refers to any of the Batch B fuel assemblies that were part of the first MP2 core. Another footnote would be added to provide an exception to LCO 3.9.19 during the initial installation of Batch B fuel assemblies in the cell blocker locations. The Action Statement would be revised to state that the provisions of TS 3.0.3 are not applicable.

1.9 TS 5.3.1, "Fuel Assemblies"

TS 5.3.1 provides design features requirements concerning fuel assemblies with respect to the reactor core. This TS currently states that reload fuel "shall have a maximum enrichment of 4.5 weight percent of U-235." This TS would be revised to state that reload fuel "shall have a maximum nominal average enrichment of 4.85 weight percent of U-235." Another sentence would be added to state that "A fuel rod shall have a maximum enrichment of 5.0 weight percent of U-235."

1.10 TS 5.6.1, "Criticality"

TS 5.6.1 provides design features requirements concerning criticality with respect to fuel storage. TS 5.6.1.a would be revised to change the allowed fuel assembly maximum nominal fuel enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for the new fuel (dry) storage racks. Another sentence would be added to state that "The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent of U-235." In addition, current TSs 5.6.1.b through 5.6.1.e would be replaced with proposed TSs 5.6.1.b through 5.6.1.h. The licensee's submittal states that the proposed wording is intended to comply with a SE issued by the Nuclear Regulatory Commission (NRC) dated October 25, 1996, that documented acceptance of topical report WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology." The submittal states that the proposed changes reflect: (1) an increase in fuel assembly maximum nominal average enrichment from 4.5 w/o U-235 to 4.85 w/o U-235, and (2) credit for soluble boron in the SFP.

1.11 TS 5.6.3, "Capacity"

TS 5.6.3 provides design features requirements regarding SFP storage capacity. The proposed change would delete an existing footnote that explains that the 1346 storage locations translate into 1306 storage locations to receive spent fuel and 40 storage locations to remain blocked. Deletion of the footnote would reflect the ability to store 40 fuel assemblies in the SFP Region B locations that have cell blockers.

1.12 TS Bases

The TS Bases would also be revised, as applicable, to address the proposed TS changes described in SE sections 1.1 through 1.11.

2.0 REGULATORY EVALUATION

Appendix A to Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), General Design Criterion (GDC) 62, “Prevention of criticality in fuel storage and handling,” states that “[c]riticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.” The NRC has established a 5% subcriticality margin (i.e., k-effective (k_{eff}) less than or equal to 0.95) for nuclear power plant licensees to comply with GDC 62.

10 CFR 50.68, “Criticality accident requirements,” states in subpart (b)(4) that “[i]f 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.”

3.0 TECHNICAL EVALUATION

The staff has reviewed the licensee’s justification for the proposed license amendment as described in Attachment 1 of the licensee’s application dated November 6, 2001, and as clarified by the supplements dated July 15, August 6, and October 29, 2002. The detailed evaluation below will support the conclusion 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.

The NRC staff has evaluated the proposed TS changes with respect to criticality considerations, boron dilution analysis, and the impact on SFP heat loads due to the proposed increased fuel storage as described in SE Sections 3.1, 3.2 and 3.3 respectively.

3.1 Criticality Considerations

3.1.1 Boraflex Degradation

As described in Section 9.8.2.1.2 of the MP2 Final Safety Analysis Report (FSAR), the SFP is located in the fuel handling area of the auxiliary building. The SFP consists of 3 regions of spent fuel storage racks, designated Regions A, B, and C. TS Figure 3.9-2 shows a schematic of the SFP layout. The SFP is designed with a maximum storage capacity of 1346 storage locations with 224 storage locations in Region A, 160 storage locations in Region B, and 962 storage locations in Region C. However, Region B currently is limited to storing 120 fuel assemblies, with 40 of the locations blocked by cell blockers in accordance with TS 3.9.19. As described in the licensee's submittal dated July 15, 2002, cell blockers serve no function other than to provide a visible cue to the fuel handler that fuel should not be inserted in that location. The cell blockers are removable, and are designed to allow fuel to be stored underneath them.

Each of the Region A and B storage locations contains 4 Boraflex panels (i.e., 384 storage locations x 4 = 1536 Boraflex panels). Boraflex is a neutron absorber material used to control reactivity in the SFP. The boraflex panels are contained inside a poison box, which is stored inside each storage cell, with the poison box itself removable from the storage cell. The Boraflex panels are sandwiched between stainless steel protective sheets.

Degradation of the Boraflex panels could result in an increase in the reactivity of the SFP configuration. Boraflex degradation has been addressed in several generic communications from the NRC staff including Generic Letter (GL) 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," which was issued on June 26, 1996. GL 96-04 addressed 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% margin to criticality could not be maintained by Boraflex material due to current or projected material degradation.

As described in the licensee's application, the criticality analysis associated with the proposed amendment takes less reactivity credit for Boraflex than is currently credited in the existing analysis. Reduced Boraflex reactivity credit in Region A and B of the SFP would be implemented to make allowance for the possibility of future degradation of the Boraflex. Reducing the amount of Boraflex reactivity credit taken would allow time to respond should Boraflex degradation be detected by the IST program.

The licensee's current criticality analysis for Region A and B credits Boraflex with the original design areal density of 0.033 +/- 0.003 grams Boron-10 (B-10) per square centimeter (cm^2). The existing analysis also assumes that 5.65 inch gaps exist in all Boraflex panels, with a random axial distribution of the gaps. The assumed criticality analysis value of 5.65 inch Boraflex gaps is based on the Electric Power Research Institute (EPRI) limiting value of 4% Boraflex shrinkage. Since the original Boraflex length in each panel is 141.25 inches, the resulting maximum axial gap would be: 141.25 inches x 0.04 = 5.65 inches.

The revised criticality analysis, performed by Westinghouse, was provided in Attachment 5 of the submittal dated November 6, 2001. The revised analysis reduces the reactivity credit for Boraflex in both Region A and B by crediting a minimum areal density of 0.025 grams B-10/cm^2. This is equivalent to stating that approximately a 25% reduction in Boraflex thickness would have to occur before impacting the criticality analysis. In addition, the Boraflex gap model has been made more conservative by changing how the gaps are axially distributed. The revised criticality analysis is similar to the existing criticality analysis, in that it assumes that 5.65 inch gaps exist in all Boraflex panels. However, the revised analysis assumes that the gaps are all lined up in 2 axial elevations, near the fuel centerline, rather than randomly distributed in the axial direction as assumed in the existing analysis. As discussed in the licensee's submittal dated July 15, 2002, the purpose of this change is to allow the possibility that future blackness testing could show a different and unexpected axial distribution of Boraflex gaps. The most recent blackness testing of the MP2 Boraflex panels found that the gaps appeared to be distributed in a generally random fashion in the axial direction (i.e., consistent with the assumption in the existing analysis). The proposed change in the gap model results in higher calculated k_{eff} values for a given set of conditions than the existing gap model since it causes more reactivity insertion due to the gap locations. This is equivalent to stating that less credit is being taken for Boraflex due to use of the proposed model. The staff concludes that the proposed gap model is conservative based on the distribution of gaps found in the most recent blackness testing.

The licensee's Boraflex IST program consists of three parts: (1) visual examination and destructive testing of the Boraflex material, (2) blackness testing, and (3) SFP silica monitoring. The three parts of the program are described in SE Sections 3.1.1.1 through 3.1.1.3, respectively.

3.1.1.1 Visual Examination and Destructive Testing

The first part of the program consists of direct visual examination and destructive testing of actual inservice Boraflex material. This is accomplished by removing a Boraflex poison box from the SFP, cutting away the stainless steel protective layer, examining the Boraflex, and then sending the selected portions of the Boraflex material offsite for testing. A visual examination and material testing is performed on the Boraflex. Material testing includes neutron attenuation testing at several locations of the Boraflex for any loss of B-10 density. The EPRI RACKLIFE model is used as a tool to help select which poison box is to be removed for testing. Boraflex poison boxes that have been removed for testing have been replaced with Boraflex poison boxes that were manufactured to the same dimensions as the original poison boxes.

In the year 2000, one poison box was removed and tested. The results of the visual examination and material testing showed that the material was in good condition. Neutron attenuation testing of the samples of the in-service Boraflex showed no detectable loss of B-10 density. Future poison box removal as part of this program is planned at approximately five-year time intervals.

3.1.1.2 Blackness Testing

The second part of the program consists of blackness testing. Blackness testing provides information on gap formation, gap distribution and gap size. The most recent blackness testing of the Boraflex panels in the MP2 SFP was performed in 1996, with the following results:

- 89 cells (356 panels) of the total of 384 boraflex cells were tested.

- 64 cells had measurable gaps and 25 cells had no detectable gaps.

- Of the 64 cells with gaps, a total of 134 gaps were measured.

- 83 of the 134 gaps were measured to be less than 1.0 inch.

- 41 of the 134 gaps were measured to be in the range of 1.0 to 1.5 inches.

- 10 of the 134 gaps were measured to be in the range 1.6 to 1.9 inches.

Distribution of Boraflex gaps was generally random in the axial direction. The largest individual gap found was 1.9 inches. A few Boraflex panels had 2 gaps in the panel, with the largest gap being a sum of 2.8 inches of total gap in that panel. This is below the 5.65 inch gap assumed in the criticality analysis.

The next blackness testing program is planned to occur when the maximum accumulated gamma dose reaches an approximate value of 3.5E10 radiation absorbed dose (rads). The most recent estimate of maximum gamma dose is about 2.3E10 rads. It is expected to take several years to reach a gamma dose of 3.5E10 rads.

3.1.1.3 SFP Silica Monitoring

The last part of the program is SFP silica monitoring. As discussed in GL 96-04, irradiated Boraflex typically contains 46% silica, 4% polydimethyl siloxane polymer and 50% boron carbide by weight. When Boraflex is subjected to gamma radiation in the SFP aqueous environment, the silicon polymer matrix becomes degraded and silica filler and boron carbide are released. The presence of silica in the SFP indicates the likely depletion of boron carbide from Boraflex. The loss of boron carbide from Boraflex is characterized by slow dissolution of the Boraflex matrix from the surface of the Boraflex and a gradual thinning of the material. In a typical SFP, the irradiated Boraflex represents a significant potential source of silica (several thousand kilograms) and is the most likely source of SFP silica contamination. The boron carbide loss can result in a significant increase in the reactivity of the storage racks.

MP2 SFP silica measurements have generally been between 1 part per million (ppm) and 3 ppm over the last 10 years. The program monitors the silica concentrations for any unusual trend. The licensee concluded that silica levels this low imply minimal Boraflex degradation. This conclusion is supported by the results from the RACKLIFE model. The MP2 RACKLIFE model is updated at regular intervals and calculates the projected amount of degradation to each Boraflex panel using input data such as gamma dose to each Boraflex panel, SFP silica measurements, various other SFP water chemistry parameters, and rack and SFP characteristics. The RACKLIFE model results show that the average Boraflex panel is reduced in B-10 areal density by about 0.8%, with a peak value in the worst Boraflex panel of about 0.9%.

3.1.1.4 Boraflex Degradation Conclusion

Based on review of the licensee's submittals and the considerations discussed in SE Sections 3.1.1, 3.1.1.1, 3.1.1.2, and 3.1.1.3, the NRC staff finds that the licensee's Boraflex IST testing program: (1) has provided data that indicate that the Boraflex material has performed acceptably to date; and (2) provides acceptable methods for assessing Boraflex degradation in the future. Therefore, the staff concludes that reducing the Boraflex reactivity credit in Regions A and B of the SFP, as proposed in the revised criticality analysis, is acceptable since there is reasonable assurance that the IST program will allow appropriate corrective actions to be taken before Boraflex degradation impacts the criticality analysis.

3.1.2 Criticality Analysis - General

The criticality analysis to support the proposed license amendment request was performed by Westinghouse and was provided in Attachment 5 of the submittal dated November 6, 2001.

The analysis was performed in accordance with the criteria of Westinghouse Owners Group Topical Report WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," which was accepted for referencing in licensing applications by the NRC in an SE dated October 25, 1996. Recently discovered limitations in the methodology described in WCAP-14416-NP-A concerning axial burnup and reactivity equivalencing have been addressed and documented in Attachment 5 of the November 6, 2001 submittal. The methodology described in WCAP-14416-NP-A ensures that k_{eff} remains less than or equal to 0.95 as recommended in ANSI/ANS-57-1983, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants," dated October 7, 1983 and in NRC guidance. The methodology also takes partial credit for soluble boron in the SFP criticality analyses and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

1) k_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level; and

2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

Westinghouse performed the criticality analysis with SCALE-PC, which includes the CSAS25 control module, which calls BONAMI, NITAWL-II and KENO-Va and employs the 44 Group ENDF/B-V cross section library. SCALE-PC was validated against 30 critical experiments. The critical experiments included 19 from the Babcock and Wilcox experiments carried out in support of the Close Proximity Storage of Power Reactor Fuel and 11 experiments from the Pacific Northwest Laboratory (PNL) experimental program. The calculations adequately reproduced the data. The same code and cross sections were used for the statistical analysis and for the calculations. The DIT code was used to generate a set of isotopic concentrations. The DIT code has been benchmarked against Combustion Engineering pressurized water reactor (PWR) cores and against other PWR lattice codes, such as CASMO, with very good agreement. The benchmarking (validation) supports the use of these codes for this application. Where fuel burnup is credited, conservative approaches were taken for axial effects (3-D calculations were used instead of 2-D calculations), and also reactivity equivalencing issues, to reflect recent NRC concerns regarding WCAP-14416-NP-A in these areas. Furthermore, criticality analysis have shown that for 0 ppm of soluble boron, under normal conditions in the SFP, the SFP would remain subcritical (i.e., k_{eff} less than 1.0), including all biases and uncertainties. The staff considers the above discussed benchmarking adequate to validate the codes for use in licensing analyses.

Westinghouse performed calculations of an infinite array of fresh and burned assemblies in a 2x2 checkerboard configuration to determine the allowable burnup/enrichment curve and the tolerance and uncertainty Δk_{eff} units. They used a full SFP model to determine the soluble boron worth and postulated accidents. The soluble boron concentration was generated to be used to determine the soluble boron needed to bring the normal condition k_{eff} down from 1 to 0.95, and to offset uncertainties.

All these codes are industry standard codes that were validated through bench-marking to relevant critical experiments. The staff has historically found these codes acceptable for licensing applications. The staff has reviewed the licensee's submittal and concludes that acceptable codes have been used to perform the criticality analysis.

3.1.2.1 Criticality Analysis - Normal Operating Conditions

The licensee performed criticality analyses for each of the regions. They performed calculations to qualify the Region A, B and C to store fresh and spent fuel assemblies. Regions A and B employ a flux trap design and a Boraflex panel in each storage cell for reactivity control. Region A and B are primarily employed for the storage of fresh unburned fuel assemblies and low burnup fuel assemblies. With no soluble boron, k_{eff} for Regions A and B was found to be 0.99173 and 0.99588 respectively, including biases and uncertainties. To bring k_{eff} down to within the NRC acceptance criteria of k_{eff} less than or equal to 0.95, both regions had to be borated with a minimum concentration of 600 ppm for normal conditions.

The licensee also performed calculations to qualify the Region C racks for the storage of spent fuel assemblies with a maximum nominal initial enrichment of 4.85 w/o U-235. These assemblies have an accumulated minimum burnup of 45.0 GWD/MTU for a fuel of initial enrichment and burnup combinations within the acceptable domain depicted in Figures 4.2-2 and 4.2-3 of Attachment 5 to the licensee's submittal dated November 6, 2001. The analyses found the k_{eff} for Region C, without boron, to be 0.99499, including biases and uncertainties, and k_{eff} is less than 0.95, with 600 ppm boron concentration in the SFP, including all the biases and uncertainties at the 95/95 level.

Fuel assemblies stored in Region C must comply with proposed TS Figures 3.9-1A or 3.9-1B to ensure that the proper burn-up has been sustained. Both burn-up and enrichments proposed in Figures 3.9-1A and 3.9-1B refer to average burn-up and enrichments.

In addition, Region C is also qualified to permit storage of CFSBs. The licensee's analysis for the CFSB, such as the burn-up versus enrichment limits, was performed assuming a very conservative k_{eff} value of 0.95, including all biases, uncertainties and zero soluble boron. The contents of the CFSBs that will be stored in Region C must comply with existing TS Figure 3.9.3.

The licensee's analysis shows that for all regions of the SFP, 600 ppm of soluble boron is needed under normal conditions to assure the k_{eff} remains less than or equal to 0.95, including biases and uncertainties. In addition, the analyses shows that even with 0 ppm of soluble boron, under normal conditions, the SFP would remain subcritical (i.e., k_{eff} is less than 1.0) including biases and uncertainties.

3.1.2.2 Criticality Analysis - Accident Conditions

The licensee's criticality analysis examined all postulated accidents that could increase the k_{eff} of the SFP. The accident event that produced the largest increase in SFP k_{eff} value was employed to determine the require soluble boron concentration necessary to mitigate this and all less severe accident events. The accident conditions examined included the following:

1) a single 4.85 w/o U-235 fresh fuel assembly is misloaded in Region A, B or C;

2) a single 4.85 w/o U-235 fresh fuel assembly is misloaded between the new fuel elevator and Region C;

3) a single 4.85 w/o U-235 fresh fuel assembly is misloaded outside of the Region A and C racks;

4) a single 4.85 w/o U-235 fresh fuel assembly is dropped on top of the fuel racks;

5) a heavy load is dropped into Region A, B or C; and

6) SFP bulk water temperature exceeds 150 °F.

The analysis determined that the limiting accident is the heavy load drop onto the Region A racks. For this case, the criticality analysis shows that an additional 800 ppm of soluble boron is needed to maintain the SFP k_{eff} less than or equal to 0.95. The total amount of soluble boron required would be 800 ppm to compensate for the reactivity increase from the heavy load drop, plus the 600 ppm for normal load conditions, for a total of 1400 ppm. This concentration of soluble boron ensures that k_{eff} is less than or equal to 0.95 including all biases and uncertainties for the heavy load drop and bounds all other less limiting accidents. Proposed TS 3.9.17 would require 1720 ppm of soluble boron at all times fuel is stored in the SFP. This value conservatively bounds the 1400 ppm required to mitigate the worst-case accident.

The licensee's analyses included the assumptions that the moderator was pure water at a temperature of 68 °F and a density of 1.0 gm/cc. The analyses also included treatment for uncertainties due to tolerances in fuel enrichment and density, storage cell inner diameter, storage cell pitch, stainless steel thickness, assembly position, calculation uncertainty and axial burnup. The licensee also appropriately determined the uncertainties at the 95/95 probability/confidence level and included a methodology bias (determined from benchmark calculations) as well as a reactivity bias to account for the effect of the normal range of SFP water temperatures.

3.1.2.3 Criticality Analysis - Conclusion

Based on review of the licensee's submittals and the considerations discussed in SE Sections 3.1.2, 3.1.2.1, and 3.1.2.2, the NRC staff finds that the criticality analysis that supports the proposed amendment was performed using acceptable codes and methods. Therefore, the staff concludes that there is reasonable assurance that the proposed amendment will result in facility operation consistent with the requirements of GDC 62 and 10 CFR 50.68 for the prevention of criticality in fuel storage and handling during normal and accident operating conditions.

3.1.3 Boron Dilution Analysis

Soluble boron is currently credited in the SFP for reactivity control only for accident conditions. Since the proposed amendment would credit soluble boron in the SFP for normal conditions, the possibility of a SFP boron dilution event must be considered.

As discussed in SE Section 3.1.2, Westinghouse Owners Group Topical Report WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," was accepted for referencing in licensing applications by the NRC in a SE dated October 25, 1996. The staff's SE stated that all licensees proposing to use the methodology described in the topical report for soluble boron credit should: (1) identify potential events that could dilute the

SFP soluble boron to the concentration required to maintain the k_{eff} limit of 0.95, (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, and (3) consider the effects of incomplete boron mixing, such as boron stratification. The SE states that this analysis should be submitted for NRC review and should be used to justify the surveillance interval used for verification of the TS minimum SFP boron concentration.

As discussed in Sections 3.1.2.1 and 3.1.2.2 of this SE, the licensee determined, through criticality analysis, a minimum acceptable boron concentration of 1400 ppm to mitigate the worst-case postulated accident and maintain the SFP k_{eff} less than or equal to 0.95. For normal conditions, the licensee determined that a boron concentration of 600 ppm is required to maintain the SFP k_{eff} less than or equal to 0.95. Proposed TS 3.9.17 would require the soluble boron concentration in the SFP to be greater than or equal to 1720 ppm at all times fuel is stored in the SFP.

Since the proposed amendment would credit a SFP boron concentration of 600 ppm for reactivity control under normal conditions, assurance must be provided that a boron dilution event will not cause SFP boron concentration to be decreased from the TS minimum value of 1720 ppm, to less than 600 ppm. The licensee's submittal states that the proposed TS required SFP soluble boron concentration of 1720 ppm was selected for the following reasons:

- It is sufficiently high to provide assurance that a postulated boron dilution event can be detected in sufficient time to detect and stop the dilution, prior to reaching 600 ppm.

- The value of 1720 ppm is consistent with existing TS 3.9.1, which requires in Mode 6 that the reactor vessel and refueling canal be maintained greater than or equal to 1720 ppm. Thus, when the transfer tube is open and the SFP is connected to the refuel pool, there will be consistent soluble boron requirements.

As discussed previously, the proposed requirement in TS 3.9.17 for a SFP soluble boron concentration of greater than or equal to 1720 ppm ensures that sufficient boron would be present to mitigate the worst-case accident conditions. However, a boron dilution event and another independent accident condition do not need to be considered to occur simultaneously according to the double-contingency principle (reference ANSI/ANS-8.1-1983). As discussed in the NRC SE dated October 25, 1996, that accepted topical report WCAP-14416-NP-A, according to this principle, it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident.

The licensee's boron dilution analysis is discussed in Attachment 1 (pages 13 through 15), and Attachment 6 of the submittal dated November 6, 2001. The analysis included the following considerations:

- There is no automatic level control system in the SFP, so that any dilution to the SFP will add water to the SFP. Therefore, the addition of unborated water to the SFP will increase the SFP level, and if not controlled, an overflow of the SFP. The analysis used the continuous feed and bleed dilution method, which assumes unborated water is being added to the SFP at a constant rate and borated water, at the current concentration, is being lost by overflow of the SFP.

- Using the feed and bleed dilution method, the volume of water needed to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm was calculated to be 230,971 gallons. Therefore, any dilution source not capable of supplying 230,971 gallons of unborated water will not be capable of diluting the SFP to 600 ppm.

- For dilution sources with automatic makeup, the capacity for dilution is essentially infinite. Should one of these sources begin adding unborated water to the SFP, the SFP level would rise to the high-level-alarm setpoint, thus alerting control room operators. Should the high-level-alarm fail, and should no plant equipment operator (PEO) actions be taken, the SFP will eventually fill to the curb and begin overflowing. The effects of the overflow would be apparent to the PEO's performing their rounds. The analysis assumes that 12 hours is the conservatively longest interval between PEO rounds to detect SFP overflow based on administrative procedure requirements. The analysis states that, assuming an arbitrary dilution flow rate of 200 gpm of unborated water, more than 19 hours are needed for the SFP soluble boron concentration to change from 1720 ppm to 600 ppm (i.e., 230,971 gallons/200 gpm = 1155 minutes = 19.25 hours). Since the conservatively longest time between PEO rounds is 12 hours, and more than 19 hours are needed at 200 gpm to dilute the SFP soluble boron concentration to 600 ppm, there is ample time to detect and secure the event.

- The licensee's analysis evaluated systems that could dilute the SFP either by direct connection to the SFP or by a potential pipe crack/break. Based on the considerations discussed above, each of the potential dilution sources was determined not to be a credible threat to dilute the SFP to 600 ppm either because: (1) the dilution source was not capable of supplying 230,971 gallons of unborated water to the SFP; or (2) for those sources having the potential to add in excess of 230,971 gallons of unborated water to the SFP, the dilution flow rate was determined to be less than 200 gpm.

- The licensee's analysis notes that the boron dilution calculations assume complete mixing of the SFP soluble boron concentration during the dilution. However, should the SFP soluble boron concentration reach 0 ppm locally around the fuel due to incomplete mixing, the SFP will still remain subcritical, (i.e., k_{eff} less than 1.0), including all biases and uncertainties.

- The licensee also considered the possibility of loss of SFP water level due to a drain-down event. The loss of water level itself will not cause a reduction in SFP boron concentration, but the subsequent recovery of SFP level could cause a reduction in the SFP boron concentration if unborated makeup water is used. There are no credible events initiated in the SFP that can cause any significant loss in SFP water level and, therefore, any make-up to the SFP, even with unborated water, would not significantly reduce the SFP soluble boron concentration. There are, however, events during refueling operations, such as a nozzle dam failure or refuel pool seal failure, which could cause significant loss of SFP level, due to the transfer tube being open to containment. If the refueling pool is flooded up and the transfer tube is open between containment and the SFP, any drain-down event that occurs in containment will affect SFP level since the transfer tube is open. It is possible that the addition of unborated water during recovery from these hypothetical events could cause significant reductions in SFP boron concentration. The licensee's submittal states that procedural controls will be implemented to the existing Abnormal Operating Procedures for these events to ensure that during the recovery from these events soluble boron concentration is maintained greater than 600 ppm.

Proposed SR 4.9.17 requires that the SFP boron concentration be verified to be greater than or equal to 1720 ppm every seven days, and within 24 hours prior to initial movement of a fuel assembly or CFSB in the SFP, or shielded cask over the cask laydown area. The licensee provided the following justification for the proposed surveillance intervals:

- No deliberate major replenishment of SFP water is expected to take place over this short period of time (seven days). Also, there is a large buffer between the minimum TS soluble boron concentration limit of 1720 ppm, and the required 1400 ppm soluble boron concentration needed for accident conditions.

- The TS require verification of the SFP boron concentration within 24 hours of fuel assembly movement, CFSB movement, or cask movement over the cask laydown area. This verifies that the boron concentration is within limits just prior to the movement.

- Any inadvertent boron dilution would then have to occur essentially concurrent to the fuel/cask movement. The hypothetical boron dilution event is independent of the fuel/cask movement and is not required to be considered by the double-contingency principle. Further, it is not credible that, while personnel are present in the SFP for fuel/cask movement, they would fail to notice SFP level increasing, an overflowing SFP, and/or significant amounts of water entering the SPF pool causing the dilution.

Based on review of the licensee's boron dilution analysis, the NRC staff finds that the licensee evaluated potential SFP boron dilution events consistent with the considerations discussed in the NRC SE dated October 25, 1996, that accepted Westinghouse Owners Group Topical Report WCAP-14416-NP-A. Therefore, the staff concludes that there is reasonable assurance that a boron dilution event would be detected and mitigated before the SFP k_{eff} exceeded 0.95.

3.1.4 New Fuel Storage Racks and Reactor Core

The proposed amendment would revise TSs 5.6.1.a and 5.3.1 to increase the allowable nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for the new fuel (dry) storage racks and the reactor core. The proposed changes would also specify that the maximum fuel rod enrichment is 5.0 w/o U-235.

As discussed on pages 9 and 10 of Attachment 1 of the licensee's submittal dated October 29, 2002, changes to TSs 5.6.1.a and 5.3.1 were submitted to the NRC on April 10, 1990, to allow enrichments up to 4.5 w/o U-235 in the new fuel storage racks and reactor core. The analysis supporting the proposed changes, performed by Advance Nuclear Fuels Corporation (ANF), evaluated the loading of fuel of enrichments up to 5.0 w/o into the new fuel storage racks. In Amendment No. 146 dated June 13, 1990, the NRC approved the proposed changes based on review of the ANF analysis. The NRC's SE for Amendment 146 indicates that the criticality analysis presented in the ANF analysis meets the acceptance criteria of maintaining k_{eff} less than or equal to 0.95.

The licensee's submittal dated October 29, 2002, states that the proposed enrichment increase, as described in the submittal dated November 6, 2001, is covered under the ANF analysis. Based on review of the licensee's submittals and Amendment No. 146, the NRC staff finds that the ANF analysis was previously accepted by the NRC and is valid for enrichments up to 5.0 w/o U-235. Therefore, the proposed changes to TSs 5.6.1.a and 5.3.1 are acceptable.

3.2 SFP Heat Loads Due to Increased Fuel Storage and Increased Enrichment

The staff reviewed the licensee's submittal with respect to the impact of the heat loads on the SFP due to the proposed change to locate fuel in the 40 Region B storage cells that are currently empty and blocked.

As discussed in the submittal dated November 6, 2001, the current MP2 design-basis SFP heat load analysis assumes that all SFP storage locations are filled with fuel at the end of plant life. The analysis assumes that all storage locations are filled by the most recently discharged fuel, thereby resulting in the largest calculated heat load. The proposed revision to LCO 3.9.19 will only allow Batch B fuel assemblies to be located in the 40 Region B blocked storage cells (fuel will be located underneath the cell blockers). As described in a new note to be added to LCO 3.9.19, a Batch B fuel assembly refers to fuel assemblies that were part of the first MP2 core. Thus, due to the longer decay time, the Batch B fuel assemblies have a smaller heat load than the fuel assumed in the heat load analysis.

Therefore, the staff finds that the current MP2 design-basis SFP heat load analysis bounds the proposed increased fuel storage since the analysis:

1) already accounts for fuel being located in the 40 additional Region B locations; and

2) uses conservative assumptions with respect to the heat loads for the additional fuel.

Based on the above evaluation, the staff concludes that the proposed change to locate fuel in the 40 Region B storage cells that are currently empty and blocked is acceptable with respect to SFP heat loads.

The submittal also states that calculations have been performed which show that the increased enrichment limit from 4.5 w/o to 4.85 w/o U-235 will not increase the previously calculated decay heat loads. Based on the licensee's statements and since decay heat load is primarily a function of the operational power and burnup, the staff concludes that the proposed increase in enrichment is acceptable with respect to SFP heat loads.

3.3 Technical Evaluation Conclusion

The NRC has reviewed the proposed TS changes and, based on the considerations discussed in SE Sections 3.1 and 3.2, concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 7414). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on 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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