Mr. Ross P. Barkhur
Vice President Operations
Entergy Operations, Inc.
Post Office Box B
Killona, LA 70066

June 14, 1995

SUBJECT: ISSUANCE OF AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. M91460)

Dear Mr. Barkhurst:

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 27, 1995.

The amendment changes the Appendix A TSs by increasing the allowable maximum enrichment for the spent fuel pool and containment temporary storage rack from 4.1 to 4.9 weight percent U-235 when fuel assemblies contain fixed poisons.

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

Sincerely,

ORIGINAL SIGNED BY:

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 108 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 14, 1995

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Vice President Operations
Entergy Operations, Inc.
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated January 27, 1995, 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. NPF-38 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 108, 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 its date of issuance, and shall be implemented within 60 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 14, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 108

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by Amendment number and contains vertical lines indicating the areas of change. The corresponding overleaf page is also provided to maintain document completeness.

REMOVE PAGE	INSERT PAGE
5-5	5-5

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a nominal total weight of 1830 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading. Assemblies shall have a maximum enrichment of 4.9 weight percent U-235 provided they contain sufficient fixed poisons to meet the final storage requirements described in Section 5.6.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full-length and 8 part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer and surge line which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,800 +600, -0 cubic feet at a nominal T_{avg} of 582.1°F.

5.5 METEOROLOGICAL TOWERS LOCATION

5.5.1 The primary and backup meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties.

b. A nominal 10.38 inch center-to-center distance between fuel assemblies placed in the spent fuel storage racks.

5.6.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation +40.0 MSL.

CAPACITY

5.6.4 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1088 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.108 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated January 27, 1995, Entergy Operations, Inc. (the licensee), submitted a request for changes to the Waterford Steam Electric Station, Unit 3 (WSES3), Technical Specifications (TSs). The requested changes would change the TSs to increase the maximum enrichment for the spent fuel pool and containment temporary storage rack from 4.1 to 4.9 weight percent (w/o) U-235 when fuel assemblies contain fixed poisons.

The staff's evaluation of the criticality aspects of the proposed changes is provided below.

2.0 EVALUATION

The WSES3 spent fuel rack is composed of cells containing stainless steel partitions which provide a fuel assembly storage area and two Boraflex insert areas per cell. The cells are oriented so that face adjacent assemblies are separated by a Boraflex insert area. The panels are arranged in a rectangular configuration which maintains a 1-inch flux trap between panels and a 10.38-inch center-to-center spacing between cells.

The NRC acceptance criterion for conforming to General Design Criterion 62 for the prevention of criticality in fuel handling and storage is that the effective multiplication (k_{eff}) of the storage racks, fully loaded with fuel of the highest anticipated enrichment and fully moderated by unborated water, shall not exceed 0.95. This value shall include all known uncertainties at the 95% probability, 95% confidence level (95/95).

The analysis of the reactivity effects of fuel storage in the spent fuel storage racks was performed with the SCALE 4 system of computer codes which includes the three-dimensional multi-group Monte Carlo computer code, KENO Va. Neutron cross sections were generated by the NITAWL and BONAMI codes. The CASMO-3 integral transport theory code was used to determine the reactivity effects of uncertainties or tolerance factors in the rack and fuel design parameters. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the WSES3 fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, absorber and assembly spacing. The intercomparison between two independent methods of analysis (KENO Va and CASMO-3) also provides an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO Va reactivity calculations, a minimum of 600,000 neutron histories were accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO Va reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the WSES3 storage racks with a high degree of confidence.

WSES3 uses the ABB Combustion Engineering 16x16 fuel rod array assembly design which contains five large water holes for control rod insertion. A typical reload fuel assembly of this design contains two separate fuel rod initial enrichments, 4.1 w/o U-235 around the water holes and assembly corners and 4.5 w/o U-235 in the remaining locations. Because of these higher fuel enrichments, reload batches usually contain burnable absorbers. The base fuel assembly used in the WSES3 fuel pool criticality analysis contained U-235 enrichments of 4.1 and 4.5 w/o and eight absorber rods (shims), each with 0.016 grams of boron-10 (B-10) per inch. These absorber rods replace fuel rods and are not susceptible to inadvertent removal. Therefore, the NRC considers them to be fixed absorbers and credit may be taken for their reactivity control effects. Various other combinations of fuel enrichments and burnable absorber loadings which meet the NRC acceptance criterion of k_{eff} no greater than 0.95 were also analyzed and the results are presented in Table 1 attached to this safety evaluation.

Uncertainties or tolerance factors in the rack and fuel design parameters were evaluated by either setting the parameter to its most adverse value or performing sensitivity studies with CASMO 3 to determine the reactivity impact of the tolerance factor. The tolerance factors included uncertainties in U-235 enrichment, fuel pellet density, fuel pellet diameter, clad I.D., guide tube thickness and burnable absorber loading. In addition, a method bias and uncertainty and an enrichment bias, determined from the benchmarking, were included, as well as the effects of Boraflex gaps as discussed below. The staff has reviewed the assumptions made in determining these biases and uncertainties and concludes that they are appropriately conservative and meet the 95/95 probability/confidence requirement.

The WSES3 Boraflex surveillance program includes periodic blackness testing, using neutron attenuation, of selected Boraflex panels in the spent fuel racks at a maximum 4-year time interval. The panels selected are those expected to receive the highest cumulative gamma dose and therefore, result in the largest gap formation. Periodic destructive testing on selected Boraflex panels will also be performed if engineering assessment determines it is necessary. Blackness testing of 697 Boraflex panels was performed in November 1992 and 538 panels were found to have no gaps. The largest gap size observed in the remaining panels was 3.6 inches, corresponding to an axial shrinkage of about 2.6%.

Since there are two Boraflex panels in the flux trap between each storage cell, it is highly unlikely that gaps would form at the same axial location in each panel. If both panels in a flux trap form gaps, but the gaps occur at different axial locations, the reactivity impact will be much lower than if the gaps occur at the same elevation. The assumption used in the WSES3 spent fuel pool criticality analysis was that all Boraflex panels contain 4.5-inch coplanar gaps at the top of each panel. This gap size bounds the WSES3 blackness measurements mentioned previously and is conservative relative to more realistic analyses based on these measurements, which would include the variations in gap size and location based on a probabilistic distribution. A licensee reactivity evaluation of placing coplanar gaps at various positions confirmed that the most reactive axial location for the placement of gaps was at the top of the panel. In addition, a 4.1% shrinkage in the width of each Boraflex panel was assumed. Therefore, the NRC staff finds these Boraflex gap assumptions acceptable.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the spent fuel racks. However, it is possible to postulate events, such as the misloading of an assembly with an enrichment and burnable absorber combination outside of the acceptable requirement, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of at least 1720 ppm of boron in the pool water required by plant procedures, since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. Therefore, the staff criterion of k_{eff} no greater than 0.95 for any postulated accident is met.

Containment temporary storage racks, which rely on fuel assembly spacing to maintain k_{eff} no greater than 0.95, are also provided in the WSES3 fuel storage facility. During normal storage conditions, assemblies in these racks are essentially neutronically decoupled due to the nominal assembly spacing of 18 inches, and k_{eff} is calculated to be less than 0.90. A dropped assembly accident was also evaluated assuming a minimum spacing of 1.762 inches between an assembly in the center rack location and the dropped assembly. In this case, credit was taken for the minimum required boron concentration during refueling (double contingency principle) and the resulting k_{eff} was also less than 0.90. These values included uncertainties and biases at the 95/95 probability/confidence level, thereby meeting the NRC acceptance criterion.

The following changes to TS 5.3.1 have been proposed as a result of the requested enrichment increase. The staff finds these changes acceptable, for the reasons stated above.

(1) The nominal total weight of uranium in a fuel rod has been increased from 1807 grams to 1830 grams.

(2) The maximum enrichment of reload fuel assemblies has been increased from 4.1 to 4.9 weight percent U-235 with the provision that the assemblies contain sufficient fixed poisons to meet the final storage requirements described in TS 5.6, i.e., the storage rack k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties.

CONCLUSION

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the WSES3 spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Although the WSES3 TSs have been modified to specify the above-mentioned fuel as acceptable for storage in the spent fuel racks, evaluations of reload core designs (using any enrichment) will, of course, be performed on a cycle by cycle basis as part of the reload safety evaluation process. Each reload design is evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and TSs to ensure that reactor operation is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21. 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on June 13, 1995 (60 FR 31171). 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.

5.0 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.

Attachment: Table 1

Principal Contributor: L. Kopp

Date: June 14, 1995

TABLE 1

NO. OF SHIMS	GM B-10/IN	ENRICHMENT	K_{EFF}
4	0.012	4.11/3.71	0.94490
4	0.016	4.20/3.80	0.94551
4	0.02	4.24/3.84	0.94757
4	0.024	4.33/3.93	0.94904
8	0.012	4.37/3.97	0.94640
8	0.016	4.50/4.10	0.94956
8	0.02	4.55/4.15	0.94706
8	0.024	4.61/4.21	0.94787
8	0.028	4.65/4.25	0.94754
12	0.02	4.85/4.45	0.94949
16	0.012	4.90/4.50	0.94598

Attachment