October 16, 1986

Docket No.: 50-382

Mr. J. G. Dewease
Senior Vice President - Nuclear Operations
Louisiana Power and Light Company
317 Baronne Street, Mail Unit 17
New Orleans, Louisiana 70160

Dear Mr. Dewease:

Subject: Issuance of Amendment No. 7 to Facility Operating License No. NPF-38 for Waterford 3

The Commission has issued the enclosed Amendment No. 7 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 25, 1986, as supplemented by letters dated August 4, 1986 and September 2, 1986.

The amendment revises the Appendix A Technical Specifications by increasing the authorized fuel enrichment limit, revising the uncertainties allowance for spent fuel storage racks, changing the surveillance interval for Control Element Assembly rod drop timing tests, correcting an error in the characterization of uranium fuel rod loading, and revising the definition of a Shift Technical Advisor.

A copy of the Safety Evaluation supporting the amendment is also enclosed.

Sincerely,

James H. Wilson, Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 7 to NPF-38
2. Safety Evaluation

cc: See next page

PBD7 JHW	PBD7	AD:PWR-B	OGC	DIR:PBD
JWilson/yt	JLee	DCrutchfield		GWKnighton
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Waterford 3

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ISSUANCE OF AMENDMENT NO. 7 TO FACILITY OPERATING
LICENSE NP. NPF-38 FOR WATERFORD 3

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

LOUISIANA POWER AND LIGHT COMPANY

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. NPF-38

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

   A. The application for amendment, dated June 25, 1986, as supplemented by letters dated August 4, 1986 and September 2, 1986, by Louisiana Power and Light Company (licensee), complies with standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 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 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. 7, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in this license. LP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James H. Wilson

James H. Wilson, Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 16, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 7

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
3/4 1-23	3/4 1-24
5-5	-
5-6	-
6-5	-
6-6	-
6-6A	-

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.0 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. T_{avg} greater than or equal to 520°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At each refueling outage.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 145 inches.

APPLICABILITY: MODES 1 and 2*#**.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 145 inches withdrawn, within 1 hour either:

a. Withdraw the CEA to greater than or equal to 145 inches, or

b. Declare the CEA inoperable and determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to greater than or equal to 145 inches withdrawn:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With K_{eff} greater than or equal to 1.0.

**Except for surveillance testing pursuant to Specification 4.1.3.1.2.

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 1807 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 and shall have a maximum enrichment of 4.0 weight percent U-235.

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.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SS	1*	1
SRO	1*	None
RO	2	1
AO	2	1
STA	1*	None

SS - Shift Supervisor with a Senior Operator License
SRO - Individual with a Senior Operator License
RO - Individual with an Operator License
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the control room command function.

*An individual with SRO/STA qualifications can satisfy the SS/STA or SRO/STA position requirements simultaneously.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Engineering and Nuclear Safety Manager.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG is an onsite independent technical review group that reports to the Engineering and Nuclear Safety Manager. The ISEG shall have the authority necessary to perform the functions and responsibilities as delineated above.

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Engineering and Nuclear Safety Manager.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall meet the requirements of either Option 1 or 2 as shown below:

a. Option 1 - Combined SRO/STA Position. This option is satisfied by assigning an individual with the following qualifications to each operating shift crew as one of the SRO's required by 10 CFR 50.54(m) (2) (i):

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*Not responsible for sign-off function.

SHIFT TECHNICAL ADVISOR (Continued)

1. Licensed as a Senior Reactor Operator on the unit and

2. Meets the STA Training Criteria of NUREG-0737, Item I.A.1.1, and one of the following educational alternatives:

   (a) Bachelor's Degree in Engineering or Science from an accredited institution;

   (b) Professional Engineers License obtained by the successful completion of the PE examination;

   (c) Bachelor's Degree in Engineering or Science Technology from an accredited institution including course work in the physical, mathematical, or engineering sciences.

b. Option 2 - Dedicated STA Position. This option is satisfied by placing on each shift a dedicated Shift Technical Advisor (STA) who meets the STA criteria of NUREG-0737, Item I.A.1.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. NPF-38

LOUISIANA POWER AND LIGHT COMPANY

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By letter dated June 24, 1986, as supplemented by letters dated August 4, 1986 and September 2, 1986, Louisiana Power and Light Company (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-38) for the Waterford Steam Electric Station, Unit 3. The proposed changes would: (1) increase the authorized fuel enrichment limit from 3.7 weight percent uranium to 4.0 weight percent, (2) revise the referenced uncertainties allowance for spent fuel storage racks; (3) change the surveillance interval for Control Element Assembly (CEA) rod drop timing tests from every 18 months to every refueling outage; (4) correct an error in the characterization of uranium fuel rod loading; and (5) revise the definition of an Shift Technical Advisor (STA).

2.0 DISCUSSION

The proposed changes to the technical specifications requested by the licensee are in five areas as described below.

2.1 Increase in fuel enrichment limit

The maximum enrichment of Waterford 3 fuel is presently limited to 3.70 weight percent U-235 by Technical Specification 5.3.1. Because Cycle 2 is being designed as an approximately 18-month cycle, increased fuel enrichments are needed. For Cycle 2 the maximum nominal enrichment will be 3.90 weight percent U-235; however, it is estimated that later cycles will require a maximum fuel enrichment of approximately 4.1 weight percent U-235. The proposed change will increase the level of enrichment for fuel to be loaded into the reactor core from a maximum of 3.70 weight percent U-235 to a maximum of 4.0 weight percent U-235. Analyses have been performed by the licensee to demonstrate the acceptability of storing fuel with a maximum enrichment of 4.1 weight percent in the fuel storage areas (spent fuel pool, new fuel storage vault, and containment temporary storage racks).

2.2 Revision to the referenced uncertainties allowance for spent fuel storage racks

The proposed change would revise Technical Specification 5.6.1, Fuel Storage Criticality by removing the numerical uncertainty associated with k-eff as well as the reference to the FSAR.

According to Technical Specification 5.6.1, k-eff must be equal to or less than 0.95 when flooded with unborated water. This includes a conservative allowance of 0.0455 delta k-eff for uncertainties as described in Section 9.1.2 and Table 9.1-8 of the FSAR. The conservative allowance value is specific to the criticality analyses currently documented in the FSAR. Middle South Services, Inc. has redone the criticality analysis for the spent fuel pool using KENO, a 3-D Monte Carlo analysis code. Based upon this analysis, the resultant k-eff is less than the required limit of 0.95 for enrichments up to 4.10 weight percent U-235.

Since the allowance for uncertainties has been changed as a result of the new analysis, it is necessary to modify the wording of this technical specification.

The reference to the FSAR is also being deleted to avoid confusion as the analysis will not be included in the Waterford FSAR until December, 1987 in accordance with 10 CFR 50.71(e).

2.3 Change to surveillance interval for CEA rod drop timing tests

The proposed change would revise Surveillance Requirement 4.1.3.4c by requiring CEA drop time measurements to be performed at each refueling outage in lieu of every 18 months. This change will accommodate the present Waterford 3 Cycle 1 length and the extended cycle lengths for Cycle 1 and subsequent cycles.

Surveillance Requirement 4.1.3.4c was originally performed on February 6, 1985. The next scheduled 18-month surveillance was due on August 6, 1986, with a late due date (as allowed by Surveillance Requirement 4.0.2a) of December 20, 1986.

Waterford 3 is currently scheduled to begin the first refueling outage in mid-November 1986 and, therefore, the late due date for Cycle 1 may be acceptable. However, should unforeseen circumstances occur, the refueling outage could be extended beyond the late due date, thus forcing a premature outage to perform Surveillance Requirement 4.1.3.4c. Additionally, Waterford 3 will be using a nominal 18-month Cycle 2 and may go to 24-month refueling cycles in the future. Even allowing the 25% interval extension from Surveillance Requirement 4.0.2a will shortly force a mid-cycle outage due to the 3.25 times restriction of Surveillance Requirement 4.0.2b.

The licensee has evaluated the factors that could potentially affect CEA drop times and has determined that these factors will not adversely change as a result of extending the test interval from 18 months to every refueling outage. These factors include changes in component clearances, changes in the physical configuration of the CEA or guide tube, and the build-up of corrosion products and suspended material in the RCS that could interfere with CEA motion.

2.4 Correction of an error in the characterization of uranium fuel rod loading

The proposed change would correct an error in Technical Specification 5.3.1 by changing the phrase "maximum total weight" to "nominal weight" in reference to the uranium loading of a fuel rod.

Combustion Engineering uses a nominal 426.55 kilograms of uranium per fuel assembly loading. Assuming no burnable poison pins (i.e., 236 fuel rods), this is equivalent (after dividing kilograms/assembly by the number of fuel rods/assembly) to a nominal 1807 grams of uranium per fuel rod. Because fuel enrichment must vary for different core regions, it is clear that referring to 1807 grams of uranium per fuel rod as the "maximum total weight" is in error.

2.5 Revision to the definition of an STA

The proposed change would revise the technical specifications on Administrative Control 6.2.4.1, Shift Technical Advisor, and Table 6.2-1, Minimum Shift Crew Composition.

On October 28, 1985 the Commission published a policy statement on Engineering Expertise on Shift (50 FR 43621). The policy statement provided two options to meet provisions of Item I.A.1.1. of "Clarification of TMI Action Plan Requirements," NUREG-0737, for an on-duty Shift Technical Advisor (STA). The first option allows a Senior Reactor Operator (SRO) to perform the dual SRO/STA function provided that the SRO meets the STA training criteria and possesses one of the educational requirements defined in the policy statement. The second option is the continued used of a dedicated STA on each shift who meets the training criteria of NUREG-0737.

The proposed change implements the combined SRO/STA position while maintaining the flexibility to use Option 2 should Option 1 requirements not be met. Specifically, Administrative Control 6.2.4.1 is rewritten to define the STA requirements consistent with the Commission policy statement. The proposed change for Table 6.2-1, which defines the minimum shift crew composition, includes a footnote to implement the dual SRO/STA position provided the SRO meets the criteria defined in Administrative Control 6.2.4.1.

3.0 EVALUATION

The proposed changes to the technical specifications requested by the licensee and described in five areas above, are evaluated below.

3.1 Increase in fuel enrichment limit

a. Analysis Methods

The analysis of the criticality aspects of the storage of Waterford 3 fuel assemblies having a fuel enrichment of 4.1 weight percent uranium-235 was performed by Middle South Services (MSS) , Inc. (Ref. 1). The FSAR analysis for the Waterford 3 fuel storage racks was for the storage of fuel assemblies having a fuel enrichment of 3.5 weight percent uranium-235 (although the Technical Specifications permit 3.7 weight percent uranium-235). The MSS analysis methods consist of the KENO-IV/S and NITAWL-S codes which are part of the SCALE-2 (Ref. 2) code package. The KENO-IV/S code is a multigroup Monte Carlo criticality program for the determination of a system's effective neutron multiplication factor (K_{eff}). This code has the capability of modeling complex, three-dimensional systems. The NITAWL-S code is used to perform the resonance self-shielding calculations for those nuclides with resonances that are important to the criticality analysis. The Nordheim (Ref. 3) integral treatment is used by NITAWL-S. A 123-group neutron cross section library was used in the criticality calculations.

The MSS analysis methods were benchmarked against critical experiments performed by Babcock & Wilcox (B&W) and described in Reference 4. Five experiments were analyzed. These experiments used arrays of UO_2 fuel rods with a fuel enrichment of 2.46 weight percent uranium-235. The results obtained indicate that the MSS analysis methods gave a mean K_{eff} of 0.9964 ± 0.0082. The mean K_{eff} obtained by MSS is in good agreement with the analysis of 21 critical assemblies by B&W (Ref. 4) which gave a mean K_{eff} of 0.9967 ± 0.0087 and the analysis of 70 critical assemblies performed by the Oak Ridge National Laboratory (ORNL) which gave a mean K_{eff} of 0.9958 ± 0.0087 (Ref. 5). The D' test (Ref. 6) was used to verify that the distribution of K_{eff}s from the three different calculations (performed by three different groups) was normal. The equivalency of the 5 MSS calculations and the combined 91 B&W and ORNL calculations were further tested with the F, t, and Bartlett tests. The result of the statistical analysis is that the one-sided, upper tolerance factor to be applied to the MSS KENO-IV/S K_{eff} calculation is 0.021 at the 95% probability at a 95% confidence level.

Although the MSS benchmarking effort is not extensive, the staff concludes that the methodology used by MSS for the analysis of the Waterford 3 fuel storage racks is sufficient and, therefore, acceptable.

b. Containment Temporary Storage Rack (CTSR)

The CTSR consists of 5 storage locations arranged in a row. The center-to-center distance between two storage locations is 18 inches with each storage location consisting of a square box with nominal internal side dimensions of 8.62 inches. Both a nominal (with the fuel assemblies centered in a storage location) and an adverse (with fuel assemblies shifted to minimize the distance to the center fuel assembly) geometry were analyzed. The adverse geometry was the most reactive. The results of the KENO-IV/S analysis for this adverse geometry flooded with pure water at a density of 1 gram per cubic centimeter are as follows:

0.8765	KENO-IV/S K_{eff}
0.0012	pellet densification factor
0.0210	one-sided, upper tolerance factor
0.8987	

Based on its review, the staff concludes that the MSS value of K_{eff} equal to 0.899 meets the staff's criterion that K_{eff} shall be less than or equal to 0.95, including uncertainties and biases at least at a 95/95 probability/confidence level, for the storage in the CTSP of fuel enriched to 4.1 weight percent in the uranium-235 isotope.

c. Fresh Fuel Storage Rack (FFSR)

The new fuel storage rack consists of an array of 8x10 storage locations. These 80 storage locations are arranged in 8 groups of 10 storage locations each. Within a group the 10 storage locations are arranged in two columns of 5 locations each and have a nominal center-to-center spacing of 21 inches between other storage locations along the rows and columns of the group. Additional spacing exists between each group in the rack.

Calculations were performed for both a nominal configuration with fuel assemblies centered in the storage locations and for two adverse geometries where the fuel assemblies were shifted within the storage locations. The KENO-IV/S results established the most reactive configuration as the nominal configuration and the most reactive moderator as full-density pure water having a density of 1 gm per cubic centimeter. For the most reactive configuration the MSS results for K_{eff} are as follows:

0.8810	KENO-IV/S K_{eff}
0.0012	pellet densification factor
0.0210	one-sided, upper tolerance factor
0.9032	

Based on its review, the staff concludes that the MSS value of K_{eff} equal to 0.903 meets the staff's criterion that K_{eff} shall be less than or equal to 0.95, including uncertainties and biases at least at a 95/95 probability/confidence level, for the fresh fuel storage racks flooded with pure water having a density of 1 gram per cubic centimeter.

The staff concludes that the licensee meets with ample margin the staff's criterion that K_{eff} shall be less than or equal to 0.98 for the case of the fresh fuel storage rack moderated by low-density, hydrogenous material since the K_{eff} was approximately 0.80 excluding uncertainties for such cases. Therefore, the staff concludes that the storage in the FFSR of fuel enriched to 4.1 weight percent in the uranium-235 isotope is acceptable.

d. Spent Fuel Storage Racks (SFSR)

The spent fuel storage rack consists of a 32x34 array of 10.329 inch (nominal) square storage locations. The interior dimension of the square storage locations is 8.57 inches (nominal). Two exterior faces of each storage location have provisions for a 0.1 inch thick sheet of Boraflex layered between two stainless steel sheets. The racks are assumed to be immersed in pure water at 20°C having a density of 1 gram per cubic centimeter.

Calculations were performed for both a nominal configuration with fuel assemblies centered in the storage locations and for an adverse configuration where the fuel assemblies were offset toward the lower right corner of each storage location. Both the nominal and adverse configurations were modeled in a simplified KENO-IV/S geometry to establish the effect on K_{eff} of non-centered fuel assemblies. The MSS results for K_{eff} for the SFSR for fuel with a 4.10 weight percent uranium-235 enrichment are as follows:

0.9177	KENO-IV/S K_{eff} (nominal configuration)
0.0012	pellet densification factor
0.0210	one-sided, upper tolerance factor
0.0001	stainless steel machining tolerance factor
0.0020	Boraflex B10 loading uncertainty factor
0.0067	adverse configuration uncertainty factor
0.9487	

Based on its review, the staff concludes that the MSS value of K_{eff} equal to 0.949 meets the staff's criterion that K_{eff} shall be less than or equal to 0.95, including uncertainties and biases at least at a 95/95 probability/confidence level, for the spent fuel storage racks flooded with pure water at a density of 1 gram per cubic centimeter.

The loss of pool cooling with a subsequent increase in the spent fuel pool temperature has been considered. The licensee states that the reactivity decreases from its nominal value with an increase in pool temperature. The licensee states that a fuel assembly drop into a horizontal position on top of the spent fuel storage rack is the most limiting accident. In this configuration the dropped assembly will be about 30 inches away from other fuel assemblies and will have a negligible reactivity effect. Moreover, for any accident conditions analyzed, credit may be taken for the soluble boron present in the spent fuel pool water which would reduce K_{eff} significantly below the criterion of 0.95. The staff concludes that credible accident configurations will not lead to a reduction in the margin to criticality for all storage racks for fuel enriched up to a maximum of 4.10 weight percent uranium-235.

e. Conclusion

Based on the review presented above, the staff concludes that the use of the spent fuel storage areas (spent fuel pool, new fuel storage vault, and containment storage racks) for storing fuel up to 4.10 weight percent U-235 is acceptable. Therefore, increasing the authorized fuel enrichment limit to 4.0 weight percent uranium-235 in Technical Specification 5.3.1 is acceptable.

3.2 Revision to the referenced uncertainties allowance for spent fuel storage racks

The licensee proposes to change Technical Specification 5.6.1a to delete reference to the allowance of 0.0455 delta K_{eff} for uncertainties and the reference to Section 9.1.2 and Table 9.1-8 of the FSAR. The new analysis methodology, as described in Reference 2, has a different uncertainty allowance for establishing the maximum K_{eff} in spent fuel pool criticality calculations. The reference to the FSAR is no longer applicable due to this change in analysis methodology. The analysis of Reference 2 will be incorporated in the FSAR in accordance with 10 CFR 50.71 (e). Therefore, the staff concludes that the proposed change to Technical Specification 5.6.1a is acceptable.

3.3 Change to surveillance interval for CEA rod drop timing tests

The licensee proposes to revise Surveillance Requirement 4.1.3.4c on control rod drop time measurements. The present Surveillance Requirement requires such measurements at least once per 18 months. The proposed change would be to perform the drop time measurements once each refueling outage. This change would conform to extended cycle operation of Waterford 3 and obviate the need for an outage to measure control rod drop times.

The control rod drop time measurements are performed for all control rods following each removal and reinstallation of the reactor vessel head and for specific control rods following maintenance or modifications to the rod drive system prior to reactor criticality. The purpose of the drop time measurements is to ensure control rod operability for reactor shutdown and to verify that the drop times conform to the transient and accident analysis reactivity insertion rate. Changes in component clearances or the physical configuration of the CEA guide tubes are most likely to occur due to reactor head removal and replacement, or maintenance on the CEA and that portion of the drive system directly interfacing with the assembly. For these cases Surveillance Requirements 4.1.3.4a and 4.1.3.4b, respectively, require CEA rod drop testing independent of the proposed change. Once control rod drop times have been measured after, for example, a refueling outage and found to be acceptable, then the drop time would be relatively unaffected by any increase in a fuel cycle's length.

Additional factors have been considered which help assure that control rod drop times have not deteriorated. Chemistry requirements (e.g., Technical Specification 3.4.6) and other controls on the reactor coolant system will minimize corrosion and the build-up of corrosion products or other suspended materials in those areas affecting CEA drop times. Each full length CEA is exercised every 31 days in accordance with Surveillance Requirement 4.1.2.1.2 and this surveillance will detect rods that are sticking. The loose parts detection system (Technical Specification 3.3.3.9) will alert the operating staff to conditions with a potential for affecting reactor internals. Additionally, planned and unplanned reactor trips could yield an indication of control rod drop time deterioration.

Based on its review review of the purpose of the control rod drop time measurements as well as factors which could affect the drop times as a function of cycle length, the staff concludes that the proposed change in Surveillance Requirement 4.1.3.4c is acceptable.

3.4 Correction of an error in the characterization of uranium fuel rod loading

The licensee proposes to revise Technical Specification 5.3.1 which presently states that a fuel rod contains a maximum total weight of 1807 grams uranium. The change would state that a fuel rod would contain a nominal total weight of 1807 grams uranium. The staff concludes that this change is acceptable because this part of Technical Specification 5.3.1 relates to the design features of a fuel assembly and the exact or maximum weight of uranium in each fuel rod is not required. Thus, the licensee's proposed change to Technical Specification 5.3.1 characterizing uranium fuel rod loading is acceptable.

3.5 Revision to the definition of an STA

The proposed change would allow Waterford Unit 3 to combine the role of the STA and an SRO, provided the SRO meets the qualifications of the STA as provided for in the Commission Policy Statement on Engineering Expertise on Shift, dated October 28, 1985. LP&L has proposed to revise Table 6.2-1 and Section 6.2.4 of the Technical Specifications to allow the use of Option 1 and Option 2 of the Commission Policy Statement.

The staff has reviewed the proposed change and finds that it conforms to the Commission Policy Statement on Engineering Expertise on Shift and is therefore, acceptable.

4.0 CONTACT WITH STATE OFFICIAL

The NRC staff has advised the Administrator, Nuclear Energy Division, Department of Environmental Quality, State of Louisiana of the proposed determination of no significant hazards consideration. No comments were received.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area. The staff has determined that the amendment involves no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazards consideration, and there has been no public comment on such findings. 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 this amendment.

6.0 CONCLUSION

Based upon our evaluation of the proposed changes to the Waterford 3 Technical Specifications, we have concluded that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable, and are hereby incorporated into the Waterford 3 Technical Specifications.

Dated: October 16, 1986

REFERENCES

1. "WSES-3 Fuel Storage Racks Upgrade For The Storage Of 4.1 Weight Percent U-235 Assemblies," M. R. Eastburn, File 304-37, Middle South Services, Inc., June 13, 1986. (Attachment C of Reference 1)

2. "SCALE-2: A Modular Code System For Performing Standardized Computer Analysis For Licensing Evaluation," NUREG/CR-0200, Oak Ridge National Laboratory.

3. L. W. Nordheim, "The Theory of Resonance Absorption," Proceedings of Symposia in Applied Mathematics, Vol. XI, 58, G. Birkhoff and E. P. Wigner, Eds., Am. Math. Soc. (1961).

4. "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," N. M. Baldwin, et al, BAW-1487-7, Babcock & Wilcox Company, July 1979.

5. "Scale System Cross Section Validation with Shipping Cask Critical Experiments," R. M. Westfall and J. R. Knight, Trans. Am. Nucl. Soc., 33, 368, 1979.

6. "Assessment of The Assumption of Normality (Employing Individual Observed Values)," American National Standard, ANS1.N15.15-1974.