November 29 1994

Mr. George A. Hunger Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: INCREASE IN SPENT FUEL POOL CAPACITY, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M88610 AND M88657)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No. 82 to Facility Operating License No. NPF-39 and Amendment No. 43 to Facility Operating License No. NPF-85 for the Limerick Generating Station (LGS), Units 1 and 2. These amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 14, 1994, as supplemented by letters dated March 22, July 14, September 1, October 21, and November 22, 1994.

These amendments revise TS sections 5.5.1.1 and 5.5.3, to permit a modification to install new high density spent fuel storage racks in each of the spent fuel pools at LGS. The new high density spent fuel storage racks will increase the spent fuel pool storage capacity in each spent fuel pool to 4117 fuel assemblies. This modification will extend the date of loss of full core discharge capability from 1998 to 2013.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. You are requested to notify the NRC when these amendments have been implemented at LGS, Units 1 and 2.

Sincerely,
/S/ J. Shea for
Frank Rinaldi, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-352/50-353

Enclosures:
1. Amendment No. 82 to License No. NPF-39
   Amendment No. 43 to License No. NPF-85
2. Safety Evaluation

cc w/encls: See next page

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NAME	MO'Brien	FRinaldi:rb	JStolz	BMB
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 29, 1994

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2. Safety Evaluation

cc w/encls: See next page

cc:

J. W. Durham, Sr., Esquire
Sr. V.P. & General Counsel
PECO Energy Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. David P. Helker, MC 62A-1
Manager-Limerick Licensing
PECO Energy Company
965 Chesterbrook Boulevard
Wayne, Pennsylvania 19087-5691

Mr. David R. Helwig, Vice President
Limerick Generating Station
Post Office Box A
Sanatoga, Pennsylvania 19464

Mr. Robert Boyce
Plant Manager
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

Regional Administrator
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Neil S. Perry
Senior Resident Inspector
US Nuclear Regulatory Commission
P. O. Box 596
Pottstown, Pennsylvania 19464

Mr. Craig L. Adams
Superintendent - Services
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

Mr. Rich R. Janati, Chief
Division of Nuclear Safety
PA Dept. of Environmental Resources
P. O. Box 8469
Harrisburg, Pennsylvania 17105-8469

Mr. James A. Muntz
Superintendent-Technical
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. James L. Kantner
Manager-Experience Assessment
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Library
US Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Larry Hopkins
Superintendent-Operations
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. John Doering, Chairman
Nuclear Review Board
PECO Energy Company
965 Chesterbrook Boulevard
Mail Code 63C-5
Wayne, Pennsylvania 19087

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 82
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated January 14, 1994, as supplemented by letters dated March 22, July 14, September 1, October 21, and November 22, 1994, 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-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 82 , are hereby incorporated into this license. Philadelphia Electric Company 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: November 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 82

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

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 area of change.

Remove	Insert
5-8	5-8

DESIGN PRESSURE AND TEMPERATURE (Continued)

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1500 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal steam dome saturation temperature of 547°F.

5.5 FUEL STORAGE

CRITICALITY

5.5.1.1 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, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal center-to-center distance between fuel assemblies placed in the storage racks of greater than or equal to 6.244 inches.

5.5.1.2 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

5.5.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 346′0″.

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4117 fuel assemblies.

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6.1-1.

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. NPF-85

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated January 14, 1994, as supplemented by letters dated March 22, July 14, September 1, October 21, and November 22, 1994, 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-85 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 43, are hereby incorporated into this license. Philadelphia Electric Company 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: November 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

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 area of change.

Remove	Insert
5-8	5-8

DESIGN PRESSURE AND TEMPERATURE (Continued)

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.

2. 1500 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal steam dome saturation temperature of 547°F.

5.5 FUEL STORAGE

CRITICALITY

5.5.1.1 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, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal center-to-center distance between fuel assemblies placed in the storage racks of greater than or equal to 6.244 inches.

5.5.1.2 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

5.5.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 346′0″.

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4117 fuel assemblies.

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 82 AND 43 TO FACILITY OPERATING

LICENSE NOS. NPF-39 AND NPF-85

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated January 14, 1994, as supplemented by letters dated March 22, July 14, September 1, October 21, and November 22, 1994, the Philadelphia Electric Company (the licensee) submitted a request for changes to the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS). The requested changes would revise TS Section 5.5.3 to allow an increase in the spent fuel storage capacity in each of the spent fuel pools to 4117 fuel assemblies. The installation of new high density spent fuel storage racks would permit the proposed increase in storage capacity. The supplemental letters provided clarifying information and corrected typographical errors in the January 14, 1994 application and did not change the initial proposed no significant hazards consideration determination.

The staff issued Amendment No. 72 to Facility Operating License No. NPF-39 for LGS, Unit 1, dated June 30, 1994, which authorized an increase from 2040 to 2500 fuel assemblies. This amendment was considered an interim amendment that allowed the transfer of all of the fuel from the Unit 2 spent fuel pool into the Unit 1 spent fuel pool, and permitted the initiation of the reracking preparatory work of the Unit 2, without any fuel in its spent fuel pool. The reracking of the Unit 1 spent fuel pool will start after completion of the reracking of the Unit 2 spent fuel pool. This will assure that the reracking of each pool will be done without any fuel in that pool during the reracking phase.

2.0 EVALUATION

Currently, each unit at LGS has its own spent fuel pool, which provides storage for new and spent fuel assemblies. The two spent fuel pools are located on a common refueling floor and are provided with interconnection for fuel transfer between the two pools. The original TS authorized a storage capacity of up to 2040 spent fuel assemblies for each spent fuel pool at the LGS facility. The staff interim TS amendment allowed an interim increase in the Unit 1 spent fuel pool from 2040 to 2500 fuel assemblies. This evaluation assesses the adequacy of increasing the capacity of each of the two spent fuel pools at LGS to 4117 spent fuel assemblies.

2.1 CRITICALITY ASPECTS

The proposed high density storage racks were designed by Holtec International and consist of an egg-crate structure with fixed neutron absorber material (Boral) positioned between the fuel assembly storage cells. The center-to-center spacing between stored fuel assemblies is reduced from 6.625 inches to 6.244 inches. The racks are designed to accommodate a standard 8x8 General Electric Company (GE) fuel assembly of 3.5 weight percent (w/o) U-235 average enrichment in the enriched zone and without credit for the gadolinium burnable poison normally contained within the fuel. Analyses of other fuel types in LGS confirmed that this is the most reactive fuel assembly.

The analysis of the reactivity effects of fuel storage in the LGS racks was performed with both the CASMO-3 two-dimensional transport theory code and the KENO-5a Monte Carlo computer code, using the 27-group SCALE cross-section library. CASMO-3 was also used to evaluate small reactivity increments associated with manufacturing tolerances. 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 LGS spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, these two independent methods of analysis (KENO-5a and CASMO-3) showed very good agreement both with experiment and with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a calculations, a minimum of 1,250,000 neutron histories in 2500 generations were accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO-5a reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the LGS storage racks with a high degree of confidence.

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Unborated pool water at the temperature yielding the highest reactivity (4°C).

(2) Assumption of infinite array (no neutron leakage) of storage cells, except for certain accident assessments.

(3) Neutron absorption effect of minor structural material is neglected.

The staff concludes that appropriately conservative assumptions were made.

For the nominal storage cell design, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, eccentric positioning, and fuel enrichment and density tolerances were accounted for. These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty were determined from benchmark calculations. The final maximum calculated reactivity resulted in a k-eff of 0.944 (CASMO-3) and 0.941 (KENO-5a) when combined with all known uncertainties. This meets the staff's criterion of k-eff no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events, such as the accidental insertion of an assembly outside and adjacent to the fuel storage rack, dropping an assembly on top of the rack, or lateral movement of the rack, which could lead to an increase in reactivity. However, such events were found to have a negligible effect and the resulting reactivity would remain below the 0.95 design basis.

The following TS changes have been proposed as a result of the requested spent fuel pool reracking. The staff finds these changes acceptable.

(1) TS 5.5.1.1 - The nominal center-to-center distance between fuel assemblies is reduced to 6.244 inches or greater.

(2) TS 5.5.3 - The maximum storage capacity of each pool is increased to 4117 fuel assemblies.

Based on the review described above, the staff finds the criticality aspects of the proposed modifications to the LGS 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.

2.2 RADIATION PROTECTION ASPECTS

2.2.1 Occupational Dose Control

The licensee estimated in its October 21, 1994, response to the staff's request for additional information (RAI) that the total occupational dose for planned reracking activities would be between 4 and 6 person-rem, per pool, including any necessary diving operations.

This overall estimate is based on individual dose estimates for each of the series of anticipated activities to be performed during the reracking operation. These activities include decontaminating the current storage racks once they are emptied and removed from the fuel pool, removing underwater appurtenances, installing new racks, and preparing the old racks for shipment.

The licensee has indicated that the movement and removal of the spent fuel pool racks will be performed using remote handling tools under continuous health physics coverage. All fuel will be removed from the Unit 2 spent fuel pool prior to divers entering the fuel pool. When reracking the Unit 1 pool, fuel may be stored in the pool, however, if diving operations are required, all appropriate precautions will be taken to ensure the divers do not approach the racks which contain spent fuel. Further, if divers are used, the licensee has committed to the guidance provided in Appendix A ("Procedures for Diving Operations in High and Very High Radiation Areas") to Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The licensee notes that detailed procedures prepared with consideration of the as low as is reasonably achievable (ALARA) principle will be utilized. In addition the licensee states in its RAI response that continuous air samplers will be utilized where a potential for significant airborne activity exists and that personnel will wear protective clothing and, as appropriate, respiratory protection equipment. The licensee further states that work activities, personnel traffic, and equipment movement will be monitored and controlled to minimize contamination and to assure that exposures are maintained ALARA.

Based on the staff's review of the licensee's application, the staff finds the proposed occupational dose control radiation protection aspects of the spent fuel pool rerack acceptable.

2.2.2 Solid Radioactive Waste

The licensee stated in its application that no significant increase in the volume of solid radioactive wastes is expected with the expanded storage capacity. During reracking operations, a small amount of additional resins may be generated by the pool cleanup system on a one time basis during the pool rack augmentation operation. Such resins would be handled in accordance with the plant's normal waste handling procedures.

Based on the staff's review, the staff finds that the licensee's plan for handling and disposing of solid radioactive waste generated in connection with the planned reracking operation meets regulatory requirements and is, therefore, acceptable.

2.2.3 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of postulated accidents and described the means for mitigation of these consequences should they occur. This evaluation included spent fuel handling accidents. Based on this analysis, the licensee concluded that the effects of the proposed TS changes are small and that the calculated consequences are within regulatory requirements and staff guideline dose values. Since the licensee proposes to utilize higher enrichment fuel, the staff reevaluated the postulated fuel handling design basis accident (DBA) for the LGS to consider the effects of more highly enriched fuel and increased fuel burnup on accident consequences.

In its evaluation for the LGS, issued on August 1983, the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses from this DBA to below the doses specified in Standard Review Plan (SRP)(NUREG-0800) Section 15.7.4.

Although the licensee did not address a specific higher fuel burnup value in its January 14, 1994, application (relative to that currently authorized), the staff evaluated the consequences of operation at a bounding value (60,000 MWD/T), because the licensee's reference to the future use of more highly enriched fuel (up to 4.9 weight percent U-235). In Table 1, the fuel handling accident doses associated with extended burnup, as well as that contained in the current licensing basis, are presented and compared to the guideline doses in SRP Section 15.7.4.

TABLE 1
Radiological Consequences of Fuel Handling Design Basis Accident (REM)

	Exclusion Area Thyroid	Low Population Zone Thyroid
Staff Evaluation August 1983	1.3	0.3
Bounding Estimates for Extended Burnup Fuel[1] (60,000 MWD/T)	1.6	0.4
Regulatory Requirement (NUREG-800, Section 15.7.4)	75	75

[1] According to NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Pacific Northwest Laboratory, 1987, increasing fuel enrichment to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/T increases the doses for fuel handling accident by a factor of 1.2.

The staff concludes that the only potential increased radiological consequences resulting from fuel handling accidents associated with extended burnup fuel are the thyroid doses. These doses remain well within the acceptance criteria given in NUREG-0800 and are, therefore, acceptable.

2.3 PLANT SYSTEMS ASPECTS

2.3.1 Spent Fuel Pool Cooling

The spent fuel pool cooling system (SFPCS) consists primarily of three heat exchangers, three pumps, and spent fuel pool (SFP) cooling water return and supply headers. The heat exchangers and pumps are arranged in parallel, normally two heat exchangers and two pumps (Pumps A and B) are in service and the third heat exchanger and pump serve as backup. Pumps A and B are powered from Class 1E sources. Heat is removed from the SFP heat exchangers by the service water system.

The SFPCS was designed to maintain the SFP water temperature at or below 140°F during normal plant operation with a normal heat load of 16.32x10^6 Btu/hr. During abnormal plant operation when a large heat load (e.g., full core off load) is placed in the SFP, a train of residual heat removal (RHR) system can be substituted for the SFPCS to cool the SFP via a cross-connection between the two systems. The RHR system has sufficient heat removal capacity to maintain the SFP water temperature at or below 140°F with a maximum anticipated heat load of 36.4x10^6 Btu/hr. The licensee has administrative controls in place to prevent the use of the cross-connection between the SFP cooling system and the RHR system unless the associated reactor is shut down and is in the refueling mode.

The licensee indicated that increasing the spent fuel storage capacity will result in a small increase (from 16.32x10^6 Btu/hr to 18.05x10^6 Btu/hr) in the maximum normal decay heat load and the peak SFP bulk temperature resulting from this heat load increase is 143°F which is slightly above the design temperature limit for the water in the SFP and the water temperature limit described in the guidance of the Standard Review Plan for SFP. However, the time period that the SFP water temperature exceeds 140°F is 2.5 days and occurs approximately 160 hours after plant shutdown. The licensee considered the slight increase (from 140°F to 143°F) in peak calculated SFP water temperature acceptable since it only occurs in a short duration during plant shutdown and that the RHR system will be available for additional SFP cooling if necessary. The licensee further indicated that the maximum decay heat load, assuming full core discharge and remaining cells filled, will increase from 36.4x10^0 Btu/hr to 37.6x10^6 Btu/hr, however, the RHR system is still capable of maintaining the SFP water temperature below 140^{0}F as described in Section 9.1.3.2.3 of the LGS Updated Final Safety Analyses Report (UFSAR).

Based on its review, the staff agrees with the licensee that the slight increase (from 140°F to 143°F) in peak calculated SFP water temperature resulting from the small increase in the maximum normal decay heat load only occurs in a short duration during plant shutdown and that the RHR system will be available for additional SFP cooling if needed. Therefore, the staff finds the above cited slight increase in peak calculated SFP water temperature acceptable.

2.3.2 Decay Heat Calculation

The licensee indicated that the above revised maximum normal and abnormal heat loads were calculated in accordance with the guidance of NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-water Reactors for Long-Term Cooling," and with the consideration of the effects of: the heat load associated with a maximum storage capacity of 4117 fuel assemblies; a 5% power uprate (increasing the rated core thermal power from 3293 MWt to 3458 MWt); a reduction in the minimum decay time until fuel movements begin; and increasing refueling cycles from 18 months to 24 months.

Based on its review of the licensee's rationale and heat load values at a similar plant (e.g., Hope Creek), the staff finds the above cited maximum normal and abnormal decay heat generated in the SFP acceptable.

2.3.3 Effects of SFP Boiling

The Licensee indicated that if there is a complete loss of capability of using SFPCS heat exchangers and RHR heat exchangers to remove heat from the SFP, then the SFP water temperature will begin to rise and eventually will reach the bulk boiling temperature. The calculated minimum time from the loss-of-pool cooling until the pool boils is 5.01 hours and the maximum boil-off rate is 80.85 gpm. However, as described in the LGS UFSAR, makeup water (e.g., non-safety related normal makeup sources or backup makeup sources via the safety related emergency service water system) can be added to maintain the pool water level.

The staff finds that cooling the SFP by allowing SFP to boil and by adding makeup water in the event of a complete loss of capability of using heat exchangers to remove heat from the SFP conforms with the guidance described in the SRP, and therefore, is acceptable.

2.3.4 Heavy Load Handling

The licensee indicated that accidental dropping of movable heavy objects into the SFP is precluded by the use of administrative procedures, electrical interlocks to limit the load travel over the SFP, and the use of guardrails and curbs around the pools and the reactor wells to prevent fuel handling and servicing equipment from falling into the pools. In addition, heavy load handling in the vicinity of the SFP is accomplished in accordance with the guidance described in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Based on its review, the staff concludes that increasing the spent fuel storage capacity from 2040 to 4117 fuel assemblies in each of the SFPs at LGS will not increase the probability of an accidental dropping of movable heavy objects into the SFP. Therefore, the staff finds that the heavy load handling at LGS is acceptable.

2.5.4 Open Issue

It should be noted that an issue associated with SFP cooling adequacy was identified in NRC Information Notice No. 93-83, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," dated October 7, 1993, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff will address this issue for the licensee as part of the generic evaluation process.

2.4 MATERIALS ASPECTS

2.4.1 Structural Materials

The licensee has selected the following structural materials for use in the proposed storage rack modification:

- American Society of Mechanical Engineers (ASME) Section II SA240-304 stainless steel for fabrication of the racks

- ASME SA240-304 for the internally threaded support legs

- ASME SA564-630 for the externally threaded support spindle - this is a precipitation hardened stainless steel, heat treated to 1100°F

- Weld material - type R308L stainless steel conforming to ASME specification SFA 5.9

ASME Section II, SA240, Type 304 stainless steel is a common austenitic alloy frequently used in nuclear applications. The choice of type 304 stainless steel for fabrication of the rack assembly legs is reasonable. The high chromium content imparts reasonable corrosion resistance to oxidizing effects of most electrolytes when at low concentration levels. The steel is, however, susceptible to corrosion in acidic solutions (pH < 7.0) containing chloride or fluoride anions. These anions can lead to pitting of the material. The corrosion effects by chloride or fluoride anions is not as pronounced in basic media (pH > 7.0).

The licensee has opted to use a Type 630 martensitic, precipitation hardened, stainless steel for the externally threaded support spindle. Type 630 stainless steel has increased strength, without suffering considerable loss of ductility. The corrosion resistance, however, is not quite as good as that of austenitic stainless steel. The Type 630 stainless steel has been heat treated at 1100°F to increase its resistance to stress corrosion cracking.

It should be noted that control of water impurities in nuclear plant SFP water is typically provided by the SFP demineralizers in the SFPCS. The demineralizers function to keep the chemistry of the SFP water approximately the same as that of the reactor coolant system, in order to minimize the probability of abnormal chemistry incursions during refueling operations when the two systems link together. Control of SFP chemistry, however, also serves to reduce corrosion effects by keeping the concentrations of water impurities at low levels. Therefore, stress corrosion cracking or pitting, induced by residual chloride or fluoride ions in the fuel pool, should not be a problem with the SA240-304 stainless steel.

2.4.2 Poison Material

The Boral panels used in the proposed rack modifications are manufactured in accordance with AAR Brooks and Perkins certified procedures. Production of Boral falls within the scope of the manufacturer's quality assurance program (10 CFR Part 50, Appendix B) for nuclear grade materials. The licensee intends to install the Boral sheets by freely inserting them between the 304 stainless steel walls of the rack assemblies and the 304 stainless steel sheaths which are to be welded to the wall.

It is evident that the insertion of the Boral panels into the sheathed areas will create a tight fit. Independent studies by industry organizations and by NRC contractors have shown that Boral may react with water or moisture to generate hydrogen gas. Production of hydrogen may result in deformation of the rack cells by imparting additional stresses on the walls. Information Notice 83-29, "Fuel Binding Caused by Fuel Bundle Deformation," was issued to alert the industry to this concern. The licensee's submittal indicates that holes at the corners of the sheath areas will create a sufficient vent path for any potential hydrogen which may be produced by a water-aluminum reaction.

The licensee has also created an accelerated Boral surveillance program to characterize the performance of the Boral panels during the remaining lifetime of the plant. This program is in accordance with the NRC Letter of April 14, 1978 to all nuclear power licensees, which stated that "Methods for verification of long-term material stability and mechanical integrity of special poison materials utilized for neutron absorption should include actual tests."

The licensee's accelerated Boral Surveillance Program calls for placing 10 Boral test coupons (mounted on a "tree") in each of the SFP rack areas of the LGS. At the end of the first five operating cycles following the modification, the coupon tree will be surrounded with eight freshly discharged fuel assemblies. This is done to assure that the coupons experience a higher radiation dose than the Boral panels in the storage racks. Beginning with the fifth spent fuel load, the fuel assemblies surrounding the test coupon tree will remain in place for the remaining life of the racks.

The accelerated Boral Surveillance Program calls for removing and testing one Boral test coupon at the following refueling outages for each unit after the rack modifications are complete: 1st, 2nd, 3rd, 5th and 8th. An additional test coupon will be removed at intervals of 5, 10, and 20 years after the eighth refueling outage. Each test panel, upon its removal, will be analyzed according to the following tests:

- Visual Observation and Photography
- Neutron Attenuation
- Dimensional Measurements (length, width, and thickness)
- Weight and Specific Gravity Analyses
- Wet Chemical Analysis (Optional)

The neutron attenuation and the dimensional measurements are the more important tests of the group since they are used to determine whether or not the coupons are exhibiting any signs of boron loss or structural deformation, respectively. The licensee's contractor has established an acceptable set of screening criteria for evaluating the Boral test coupons. The results of testing on the Boral test coupons will be compared to identical tests run on the Boral control coupons.

2.4.3 Conclusion

The license amendment submittal indicates that material selection for the LGS spent fuel rack modifications have been satisfactorily thought out. The racks are to be constructed from a Type 304 stainless steel fabricated according to an approved ASME Section II specification. Boral is an acceptable poison material; however, since the Boral may generate hydrogen when in contact with water or moisture, care must be taken to provide a sufficient path to allow potential hydrogen generation to vent from the sheath area. The Boral surveillance program will provide a reliable method of assessing the potential deformation or degradation of Boral panels which are exposed to radiation in the spent fuel area over time. Following the review of the licensee's submittal, the staff concludes that the licensee's selection of structural, welding and poison materials meets current industry and regulatory standards and that these materials are acceptable for construction of the new rack modules.

2.5 STRUCTURAL ASPECTS

2.5.1 High Density Racks and Rack Overhead Platforms

The high density spent fuel storage racks are seismic Category I equipment, and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee used a computer program, DYNARACK, for dynamic analysis to demonstrate the structural adequacy of the LGS spent fuel rack design under the earthquake loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor of the storage pool. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements as defined in the program was used to simulate three dimensional dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Two model analyses were performed: the 3-D single rack model analysis and the 3-D whole pool multi-rack (WPMR) analysis. The main purpose of the WPMR analysis was to investigate the fluid-structure interaction effects between racks and pool walls as well as those among the racks. Three cases of the WPMR analysis were considered: (1) 14 new racks and 1 existing rack, (2) 15 racks without overhead platform, and (3) 15 racks with 2 overhead platforms on the B3 and E2 racks, respectively, as shown in Figure 6.1.1 of the initial submittal. For the 3-D single rack model analysis, two rack geometries were considered for the calculation of stresses and displacements: (1) 18 ft (W) x 18 ft (L) x 15 ft (H), and (2) 14 ft (W) x 18 (L) ft x 15 ft (H), where W, L and H are defined as width, length and height of a rack, respectively. Each rack was considered fully loaded, partially loaded, and almost empty with two different coefficients of friction between the rack and the pool floor (μ=0.2 and 0.8) to identify the worst case response for rack movement and for rack member stresses and strains.

The seismic analyses were performed utilizing the direct integration time-history method. Four sets of time histories were calculated for each of the dynamic events (SSE, operating basis earthquake (OBE), safety relief valve (SRV) load and loss of coolant accident (LOCA) load) from the plant response spectra as described in the LGS FSAR. Among the four sets of the time histories, a controlling set for each dynamic event was chosen as that set which results in maximum stress and strains at the top of the rack.

A total of 62 3-D single rack model analyses were performed. The calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code (1986 edition), Section III, Subsection NF as shown in Tables 6.7.2-6.7.64 of the initial submittal. The results show that all induced stresses under the load combinations (Levels A, B and D service limits) are smaller than the corresponding allowable stresses specified in the code indicating that the rack design is adequate.

The licensee also calculated the weld stresses of the rack under the dynamic loading conditions. Three weld locations were considered: (1) baseplate-to-rack, (2) baseplate-to-pedestal, and (3) cell-to-cell connections. Table 6.7.27 of the initial submittal shows the ratio of the calculated weld stress with respect to the allowable stresses specified in ASME Code Section III, Subsection NF. The calculated factors of safety are in the range of 3.2 to 4.3 indicating that the weld connection design of the rack is adequate.

In the 3-D whole pool multi-rack (WPMR) analyses, all racks were considered fully loaded, and they were subjected to the dynamic loading conditions (Level D service limit). The results of the multi-rack analysis indicate that the calculated stresses on a rack are higher than those obtained from the corresponding single rack analysis. However, all calculated stresses for the multi-rack analysis are smaller than the allowable stress of the ASME Code, Section III, Subsection NF.

The results of the multi-rack analysis, however, show that the calculated maximum horizontal displacements at the top and baseplate levels of the rack are about 4.874 inches and 4.634 inches, respectively. These computed horizontal rack displacements show that rack-to-wall impacts will not occur, but rack-to-rack impacts will occur during a seismic (SSE) event. Accordingly, the licensee designed the racks with the baseplate extensions and the top bumper bars so that rack-to-rack impacts should be minimized. An impact analysis was performed as shown in Table 6.8.1 of the initial submittal. The results show a large margin of safety indicating a conservatism in the rack design.

The staff also reviewed the structural adequacy of two overhead platforms for emplacement on two rack modules, B3 and E2, respectively, as shown in page 2.1, Figures 2.1.3 through 2.1.6, and Figure 6.1.1 of the initial submittal. Although the licensee does not have an immediate, specific plan to install such a structure at the LGS, the licensee is considering this as a future option. The staff has reviewed the structural adequacy of the overhead platform as part of this evaluation.

The licensee used a computer program, ANSYS, for dynamic analysis to demonstrate the structural adequacy of both platform designs under seismic loading conditions. The platform is considered as a seismic Category I structure, and is designed to hold 12,000 lbs. of uniformly distributed dead load. The September 1, 1994, submittal shows the results of the platform analysis that the stresses in the platform remain below the allowable stresses of the ASME Code under normal condition (Level A service limit) and an SSE event (Level D service limit). It also shows that the platform will not separate from the rack during the seismic event with a large margin of safety.

Based on: (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) the large factor of safety of the induced stresses and strains of the rack when they are compared to the corresponding allowables provided in the ASME Code, Section III, and (3) the licensee's overall structural integrity conclusions supported by both single analyses and multi-rack analyses including the overhead platforms, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions, and, therefore, are acceptable.

However, it is quite likely that the racks will move during or after seismic events. Therefore, the licensee is required to institute a surveillance program that inspects and maintains the originally installed rack gaps after the occurrence of an earthquake equivalent to or larger than an OBE, if any occurs. In addition, if the licensee installs a platform structure at either LGS Units 1 or 2, the licensee is required to adhere to the following conditions: (1) the platform emplacement on only two racks modules, B3 and E2, (2) the maximum of 12,000 lbs. uniformly distributed dead load on each platform, (3) no separation and slip of a dead load from the platform during a dynamic event by providing proper anchorage, (4) the maximum allowable centroid of the platform and dead load is 20 inches above the top of the platform, and (5) further NRC review and approval on the platform analysis and design if there is any change. Furthermore, the licensee shall implement its quality assurance and inspection programs.

2.5.2 Spent Fuel Storage Pool

The spent fuel pool structure is a reinforced concrete structure and is designed as a seismic Category I structure. The dimension of the LGS pool structure is approximately 26 feet wide and 45 feet long with 6 feet thick reinforced concrete. The internal surface of the pool structure is lined with stainless steel to ensure water tight integrity.

The pool structure was analyzed by using the finite element computer program, ANSYS, to demonstrate the adequacy of the pool structure under fully loaded high density fuel racks with all storage locations occupied by fuel assemblies. The fully loaded pool structure was subjected to the load combinations specified in Tables 3.8-9, 3A-14, and 3A-15 of the LGS FSAR, including thermal loadings. Table 8.2 of in the initial submittal shows the predicted factors of safety varying from 1.05 to 1.32 for bending moments of the concrete walls and slab.

In view of the calculated factors of safety, the staff concludes that the pool structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading, and SSE loading conditions. Thus, the staff finds the storage fuel pool design acceptable.

2.5.3 Fuel Handling Accident

The following two refueling accident cases were evaluated by the licensee: (1) drop of a fuel assembly with handling equipment, which enters an empty cell and impacts the baseplate (deep drop scenario), and (2) drop of a fuel assembly with a channel, which impacts the top of a rack (shallow drop scenario).

The analysis results of Accident Case (1) shows that the load transmitted to the liner through the structure is properly distributed through the bearing pads located near the fuel handling area, therefore, the liner would not be damaged by the impact. The analysis results of Accident Drop Case (2) shows that damage will be restricted to a depth of 2.29 inches below the top of the rack, which is above the active fuel region.

The staff has reviewed the analysis results submitted by the licensee and concurs with its findings. They are acceptable based on the structural integrity conclusions supported by the parametric studies.

2.5.4 Conclusion

Based on the review and evaluation of the the licensee's initial submittal, and the additional information and analysis provided by the licensee, the staff concludes that the structural analysis and design of the spent fuel rack modules, including overhead platforms and the SFP structure, are adequate to withstand the effects of the required loads. The analysis and design are in compliance with current licensing basis set forth in the FSAR and applicable provisions of the Standard Review Plan (SRP), therefore, they are acceptable provided that the licensee commits (1) to implement a surveillance program that inspects and maintains the originally installed rack gaps after the occurrence of an earthquake equivalent to or larger than an OBE, and (2) to adhere to the conditions of staff acceptance of the platform design as indicated in Section 2.5.1.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. 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 have been prepared and published in the Federal Register on November 23, 1994 (59 FR 60376). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: F. Rinaldi
L. Kopp
D. Carter
J. Minns
D. Shum
J. Medoff
Y. Kim

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