August 16, 1990

Docket Nos. 50-352
and 50-353

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SDembek		

Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: REFUELING PLATFORM MAIN AND AUXILIARY HOISTS SURVEILLANCE REQUIREMENTS, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. 76996 AND 76997)

The Commission has issued the enclosed Amendment No. 43 to Facility Operating License No. NPF-39 and Amendment No. 8 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 22, 1990.

These amendments revise the Surveillance Requirements of the TSs for the refueling platform main and auxiliary hoists to more accurately reflect their actual use.

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

Sincerely,

Original signed by
Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 43 to License No. NPF-39
   Amendment No. 8 to License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

CP 1

LI 1 & 2 TAC 76996/7

PDI-2/LA	PDI-2/PE	PDI-2/PM	C/PRPB	OGC	PDI-2/D
MO'Brien	SDembek	RClark	LCunningham	EHollen	WButler
7/23/90	7/23/90	07/23/90	7/26/90	8/13/90	8/15/90

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

August 16, 1990

Docket Nos. 50-352
and 50-353

Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
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SUBJECT: REFUELING PLATFORM MAIN AND AUXILIARY HOISTS SURVEILLANCE REQUIREMENTS, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. 76996 AND 76997)

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

cc w/enclosures:
See next page

Mr. George A. Hunger, Jr.
Philadelphia Electric Company

Limerick Generating Station
Units 1 & 2

cc:

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
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 June 22, 1990, 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. 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

Original signed by

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1990

PDI-2/LA	PDI-2/PE	PDI-2/PM	C/PRPB	OGC	PDI-2/D
MO'Brien	SDembek	RClark	LCunningham	EHoller	WButler
7/23/90	7/23/90	07/23/90	7/26/90	8/13/90	8/15/90

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
3/4 9-7	3/4 9-7*
3/4 9-8	3/4 9-8
3/4 9-9	3/4 9-9
3/4 9-10	3/4 9-10*

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of incore instrumentation and control rods with their normal drive system.

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling platform main hoist used for handling of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1150 ± 50 pounds.

b. Demonstrating operation of the hoist loaded control rod block interlock on the main hoist when the load exceeds 485 ± 50 pounds.

c. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 + 0, - 115 pounds.

d. Demonstrating operation of the uptravel interlock when uptravel brings the top of the active fuel to not less than 8 feet 0 inches below the normal water level.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.6.2 The refueling platform frame-mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the frame mounted hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the frame mounted hoist when uptravel brings the top of a control rod to not less than 6 feet 6 inches below the normal fuel storage pool water level.

4.9.6.3 The refueling platform monorail mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the monorail hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the monorail hoist when uptravel brings the top of a control rod to not less than 6 feet 6 inches below the normal fuel storage pool water level.

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks which prevent crane travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation.

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
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 June 22, 1990, 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. 8 , 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

Original signed by

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1990

PDI-2/LA	PDI-2/PE	PDI-2/PM	CXPRPB	OGC	PDI-2/D
MO'Brien	SDembek	RClark	LCunningham	E Holler	WButler
7/23/90	7/23/90	07/23/90	7/26/90	8/2/90	8/15/90

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 8

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
3/4 9-7	3/4 9-7*
3/4 9-8	3/4 9-8
3/4 9-9	3/4 9-9
3/4 9-10	3/4 9-10*

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of incore instrumentation and control rods with their normal drive system.

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling platform main hoist used for handling of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1150 ± 50 pounds.

b. Demonstrating operation of the hoist loaded control rod block interlock on the main hoist when the load exceeds 485 ± 50 pounds.

c. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 + 0, - 115 pounds.

d. Demonstrating operation of the uptravel interlock when uptravel brings the top of the active fuel to not less than 8 feet 0 inches below the normal water level.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.6.2 The refueling platform frame-mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the frame mounted hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the frame mounted hoist when uptravel brings the top of a control rod to not less than 6 feet 6 inches below the normal fuel storage pool water level.

4.9.6.3 The refueling platform monorail mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the monorail hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the monorail hoist when uptravel brings the top of a control rod to not less than 6 feet 6 inches below the normal fuel storage pool water level.

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks which prevent crane travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 43 AND 8 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 June 22, 1990, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2. These proposed amendments would revise the Surveillance Requirements (SR) of the Technical Specifications (TSs) for the refueling platform main and auxiliary hoists to more accurately reflect their actual use. The following provides a general description of the refueling platform and associated hoists and the proposed changes for the SRs.

The refueling platform is used to transport fuel and reactor core components to and from the refuel floor pools and cavities, and as a work platform from which underwater activities can be conducted. The refueling platform has three hoists through which many of these activities are accomplished. The main hoist assembly is suspended from a trolley system on the forward side of the platform and is used for transporting and orientating fuel assemblies and control rod guides for reactor, storage rack, and shipping cask (fuel assemblies only) placement. Two auxiliary hoists, each with a 1000 pound operating capacity are provided on either side of the platform. These hoists are used to perform non-fuel core component activities involving core power monitors, control rods, control rod guide tubes, fuel support casting, neutron source holders, and general servicing aids.

Procedurally, the main hoist is required to be used for the handling of the fuel assemblies or control rod guide tubes. During the transfer of fuel assemblies and double control rod guide tubes between the reactor vessel and the spent fuel pool, a potential currently exists for component contact with pool/cavity structures (e.g., portable refueling shield) due to lack of clearance. This could cause equipment and/or carried component damage.

Therefore, the licensee proposes to change the SRs for the main hoist to allow the normal up stop limit switch to be repositioned no more than 6 inches higher to provide more clearance between a main hoist grapple-carried component and pool/cavity structures. This will maintain not less than 8 feet 0 inches of water over the top of active fuel with the pools at normal water level, which will correspond to approximately 6 feet 6 inches of water above the top of the carried fuel assembly. Also, the licensee proposed to clarify the main hoist SRs to remove the reference to control rods, since the main hoist is not used for the handling of control rods and add the phrase "not less than" before the uptravel stop distance.

Also, the proposed TS changes will remove the requirement for a fuel loaded auxiliary hoist interlock by prohibiting the lifting of a fuel assembly with the auxiliary hoist, and also permit less water above the top of a carried component. Part of the proposed auxiliary hoist TS change will clarify the requirements by adding the phrase "not less than" before the uptravel stop distance.

2.0 EVALUATION

When handling irradiated fuel, the radiation dose rates external to the pool surface are highly dependent upon the time interval from reactor shutdown. TS Section 3.9.4, "Decay Time," requires the reactor to be subcritical for at least 24 hours prior to movement of irradiated fuel in the reactor. This requirement ensures sufficient time has elapsed to allow for radioactive decay of the short lived fission products. With the pools at normal water level, the proposed six (6) inch reduction in water shielding (for the main hoist SR) in combination with the handling of a spent fuel assembly 24 hours after reactor shutdown, would raise expected radiation dose rate levels at the pool surface from 10.6 millirem/hour to 24 millirem/hour. The higher radiation dose rate is still well within the radiation zone designation for the refuel floor pool area (Radiation Zone IV, i.e., <100 millirem/hour, 24 hours after reactor shutdown).

Due to the complexity of the activities required to be accomplished to ready the refuel floor and equipment for core component handling, fuel assembly transfer within six (6) days of reactor shutdown is unlikely. To conservatively estimate pool surface radiation dose rates during fuel handling activities, core off-load and subsequent reload were assumed to occur on the third and 30th day, respectively, after reactor shutdown. The estimated pool surface radiation dose rates for a spent fuel assembly having 8 feet 0 inches of water above the top of active fuel three (3) days and 30 days after reactor shutdown are 18.0 millirem/hour (an increase of 10.3 millirem/hour from the 8 feet 6 inch water shielding condition), and 4.3 millirem/hour (an increase of 2.4 millirem/hour from the 8 feet 6 inch water shielded condition). The increased radiation levels will be limited to the transfer time between the reactor and the spent fuel pool, which is typically not more than four (4) minutes. Therefore, with the pools at normal water level, the increase in received dose to an individual per transfer to the spent fuel pool from the vessel three (3) days after shutdown, and to the vessel from the spent fuel pool 30 days after shutdown, would be approximately 0.69 millirem and 0.16 millirem respectively.

A complete core off-load and reload consists of approximately 1528 fuel assembly transfers, including 300 transfers of non-irradiated fuel. The total increase in radiation dose received by two individuals on the refueling platform with the pools at normal water level during a complete core off-load and reload is estimated to be 1.212 rem. Although this potential increase in received radiation dose would be notable relative to the past refuel floor outage man-rem total of 32.5 (representing approximately a 3.7 percent increase), it will be insignificant relative to the total outage man-rem of 209.5 (representing approximately a 0.6 percent increase). The actual received dose could be less than this value since fuel handling is not anticipated to occur before the sixth day after shutdown and approximately two thirds of the fuel will have been irradiated one or two operating cycles. Both of these factors will reduce the actual radiation levels external to the pool surface and subsequently the accumulated dose.

During a refueling outage when fuel assemblies will be shuffled in the core, approximately one third of the core will be off-loaded. The total increase in radiation dose received by two individuals on the refueling platform with the pools at normal water level during a core shuffle is estimated to be 372.6 millirem. This increase is insignificant relative to both the refuel floor outage man-rem total (approximately a 1.2 percent increase), and the total outage man-rem (approximately a 0.2 percent increase). Again, this estimate is higher than that which would actually be received due to the conservatism of using radiation dose rate levels for fuel moves three (3) days after reactor shutdown vice six (6) days.

The fuel handling accident is discussed in FSAR Section 15.7.4. The accident is assumed to occur as a result of a failure of the fuel assembly lifting mechanism resulting in dropping a raised fuel assembly onto other fuel assemblies. The accident scenario that produces the largest number of failed fuel rods is the drop of a fuel assembly and grapple mast assembly into the reactor. The analysis of this scenario revealed that the calculated exposures for the design basis accident are well within the guidelines of 10 CFR 100. A fuel assembly weighing 700 pounds was assumed to drop 32 feet and the grapple mast assembly weighing 500 pounds was assumed to drop 47 feet. The energy available for fuel damage from these objects was calculated to be 45,900 foot-pounds.

Allowing a fuel assembly to be raised to a higher elevation over the reactor will make more energy available for fuel damage than that which is currently available. The drop distances used in the analysis represent the differences in plant elevation from both the lowest point on a carried fuel assembly and the lower surface on the grapple head to the upper channel surface of fuel in the reactor. The carried fuel assembly is at a plant elevation where the water above top of active fuel will be changed from 8 feet 6 inches to 8 feet 0 inches (referenced to pool normal water level). The possible drop distances for the fuel and grapple mast assemblies will increase from not more than 31 feet 5 inches and 46 feet 0 inches to not more than 31 feet 11 inches and 46 feet 6 inches, respectively. The energy which would be available to cause fuel damage associated with these higher drop distances would increase from its present calculated value of 44,994 foot-pounds to 45,594 foot-pounds. However, as stated above, the fuel handling accident scenario discussed in the FSAR assumed an even greater drop distance and resulting energy availability with which to cause fuel damage. Therefore, the analysis will continue to bound any possible main hoist component drop scenario.

The proposed changes to the main hoist TS SRs will not result in any physical changes to the refueling platform other than the relocation of the main hoist normal up limit switch. The limit switch will be relocated on the main hoist grapple mast such that main hoist motion will stop not higher than six (6) inches from its current position. The limit switch will be reattached to the mast in a manner similar to that which was originally done. No refueling platform control logic circuits will be altered. The handling of fuel and other core components and the performance of other underwater activities will not be performed differently from previous refueling activities. Administrative controls will not be modified to accommodate these proposed changes.

Since the auxiliary hoists are procedurally prohibited from handling fuel, all normal auxiliary hoist activities will involve hoist loads less than 300 pounds. Therefore, no need exists for the auxiliary hoists to have fuel associated load and interlock capabilities. The heaviest core component normally handled by an auxiliary hoist is the control rod guide tube. During the handling of a control rod guide tube, the hoist load will be no greater than 292 pounds (i.e., control rod guide tube weight 257 pounds (dry), control rod tube grapple weight 35 pounds (dry)). Since the auxiliary hoists are prohibited from handling fuel assemblies (channeled fuel assembly weight 682 pounds (dry)), a 1000 pound capacity is not needed. Therefore, restricting hoist loads to 500 ± 50 pounds will have no adverse effect on normal hoist operation. The 500 ± 50 pound limit is consistent with and will enforce the administrative controls already in place to prevent using an auxiliary hoist to move fuel.

The 400 ± 50 pound auxiliary hoist fuel-loaded signal provides input to the refuel interlock circuitry to indicate an auxiliary hoist on the refueling platform is loaded with a fuel bundle. Several interlocks are associated with this feature and result in the following:

1. Prevention of travel of the refueling platform over the reactor while a control rod is withdrawn and any hoist is fuel-loaded.

2. Prevention of lifting of a fuel assembly from the reactor with a control rod withdrawn.

3. Prevention of withdrawal of a control rod blade with the refueling platform over the reactor and any hoist fuel-loaded.

Since the licensee is proposing to limit the auxiliary hoists' capacity to 500 ± 50 pounds, thereby precluding the use of the auxiliary hoists to move fuel, imposition of a 400 ± 50 pound fuel-loaded interlock on the auxiliary hoists is unnecessary. In addition, since the auxiliary hoists are prohibited from handling fuel, specifying the minimum water depth reference requirement to the top of active fuel (i.e., 8 feet 6 inches below normal water level) for control rod blade handling is inappropriate. Minimum water depth requirements for the auxiliary hoists need to be specified such that the reference will be consistent with the use to which the hoist will be subjected. The reference that will be used is not less than 6 feet 6 inches of water above any carried component. This will allow unimposed passage of all major core components through the spent fuel pool to the reactor well canal, while maintaining adequate shielding for the irradiated components being handled. Currently, during the transfer of core components between the reactor vessel and the spent fuel pool, the potential exists for component contact with pool/cavity structures (e.g., portable refueling shield) due to lack of clearance. This could cause equipment and/or carried component damage. A control rod blade, one of the larger core components, during transfer from the reactor to the spent fuel, will have approximately six (6) inches of clearance between the bottom of the blade and the floor of the shield bridge in the canal upon implementation of the proposed normal up stop limit.

Permitting non-fuel core components to be raised to a higher plant elevation than previously allowed will increase the radiation levels external to the pool surface. The control rod blade will create the greatest radiation hazard during handling. Currently, 7 feet 0 inches of water shielding are provided as described in FSAR Section 9.1.4.3. The calculated average surface radiation dose rates with 7 feet 0 inches and 6 feet 6 inches of water shielding are 10.0 millirem/hour and 27.0 millirem/hour, respectively. The maximum calculated surface radiation dose rates considering worst case component material compositions would be 20.0 millirem/hour and 54.0 millirem/hour, respectively. These higher possible radiation dose rates are still well within the radiation zone designation for the refuel floor pool area (Radiation Zone IV, i.e., <100 millirem/hour, 24 hours after reactor shutdown). A six (6) inch reduction in water shielding will increase the calculated radiation dose rates by a factor of approximately 2.5. This increase in radiation levels will be limited to the transfer time between the reactor and the spent fuel pool, which is typically not more than five (5) minutes. Therefore, with the pools at normal water level, the increase in received dose to an individual would average approximately 1.42 millirem per component transferred. During an outage, vessel to spent fuel pool control rod blade transfers should not normally exceed 30. The increased radiation dose received by two individuals on the refueling platform with the pools at normal water level is estimated to average 85.2 millirem. This potential increase in received radiation dose will be insignificant relative to the past refuel floor outage man-rem total of 32.5 (representing approximately a 0.26 percent increase), and past total outage man-rem of 209.5 (representing approximately a 0.04 percent increase). The actual radiation levels and total received dose are expected to be less than those predicted since the control rod blade design providing the above estimates was an advanced type (General Electric DURALIFE 215) not currently in use at LGS. The advanced type of control rod blade has a longer in-vessel life than those currently in use, and therefore would become more activated than the control rod blades currently in use.

The fuel handling accident discussed in FSAR Section 15.7.4 and summarized above is also pertinent to the safety discussion concerning the auxiliary hoists.

Allowing a core component to be raised to a higher plant elevation over the reactor vessel will increase the potential energy available for fuel damage provided the drop weight is maintained the same. The greatest possible distance through which an object (assumed to be at least one (1) foot long) could drop would increase from not more than 45 feet 1 inch to not more than 45 feet 7 inches, a 1.1 percent increase in the drop distance. However, since auxiliary hoist load will be restricted to 500 ± 50 pounds, half of the currently allowed limit, the energy which would be available to cause fuel damage would decrease from an approximate value of 45,000 foot-pounds to 22,790 foot-pounds. Therefore, all auxiliary hoist component drop scenarios possible will continue to be bounded by the current analyses.

A control rod removal error during refueling activities is discussed in FSAR Section 15.4.1.1. The transient considered was an inadvertent critically due to the complete withdrawal or removel of the highest worth control rod during refueling. However, the core is designed to remain subcritical and meet shutdown requirements with the highest worth rod withdrawn. During refueling operations, system interlocks are provided to assure that inadvertent criticality does not occur because two control rods have been removed or withdrawn together. Refueling interlocks are provided to accomplish the following.

1. Prevent refueling platform travel over the reactor core if a control rod is withdrawn and fuel is on the hoist.

2. Prevent control rod motion if the refueling platform is over the reactor core and fuel is on the hoist.

These interlocks back up requirements that all control rods be fully inserted when fuel is being loaded into the core. Another interlock that is provided involves the reactor mode switch. With the mode switch in the "Refuel" position, only one control rod can be withdrawn at a time. Finally, the design of the control rod blade does not physically permit its removal from the reactor since the fuel support piece and control blade are designed so that the blade can not be removed from the reactor without prior removal of the four adjacent fuel assemblies. The withdrawal of the highest worth control rod during refueling will not result in criticality and additional reactivity insertion is precluded by interlocks and physical design.

The proposed changes to the TS SRs on the auxiliary hoists will not result in any physical changes to the refueling platform or its auxiliary hoists. This change will not alter the physical load capacity of the auxiliary hoists since no material changes are being performed and the hoists will be maintained in the same manner. No refueling platform control logic circuits will be altered. The handling of core components and performance of other underwater activities will not be performed differently from previous refueling activities. Administrative controls will not be modified to accomodate these changes.

We have reviewed the licensee's analyses and agree with their evaluations. We conclude that the proposed changes to the SRs for the main and auxiliary hoists will result in minimal increases in occupational radiological exposures, the applicable design analyses remain bounding and all regulatory requirements will continue to be met so that the proposed changes will not adversely affect safety. The proposed changes to the SRs of the TSs are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet 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 nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register (55 FR 28479) on July 11, 1990 and consulted with the Commonwealth of Pennsylvania. No public comments were received and the Commonwealth of Pennsylvania did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security nor to the health and safety of the public.

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