June 17, 1999

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
P. O. Box 63
Lycoming, NY 13093

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 TO REFLECT A PLANNED MODIFICATION TO INCREASE THE STORAGE CAPACITY OF THE SPENT FUEL POOL (TAC NO. MA1945)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 15, 1998, as supplemented by letters dated September 25, October 13, December 9 (two letters), 1998; January 11, April 1, and April 22, 1999.

This amendment changes Technical Specification (TS) 5.5, "Storage of Unirradiated and Spent Fuel," to reflect a planned modification to increase the storage capacity of the spent fuel pool from 2776 to 4086 fuel assemblies. It also deletes an inappropriate statement and reference within TS 5.5.

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

Sincerely,

Original signed by:

Darl S. Hood, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000220
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Docket No. 50-220

Enclosures: 1. Amendment No.167 to DRP-63
2. Safety Evaluation

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DATED: June 17, 1999

AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DRP-63 NINE MILE POINT NUCLEAR POWER STATION UNIT NO. 1

Docket File
PUBLIC
PDI-1 Reading
S. Bajwa
S. Little
D. Hood
OGC
G. Hill (2), T-5 C3
W. Beckner
ACRS
M. Oprendek, Region I
R. Scholl (e-mail SE only to RFS)
L. Kopp
C. Hinson
D. Shum
K. Parczewski
Y. Kim
B. Thomas
C. Lauron

cc: Plant Service list

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Sincerely,

Original signed by:

Darl S. Hood, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No.167 to DRP-63
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 17, 1999

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
P. O. Box 63
Lycoming, NY 13093

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 TO REFLECT A PLANNED MODIFICATION TO INCREASE THE STORAGE CAPACITY OF THE SPENT FUEL POOL (TAC NO. MA1945)

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Darl S. Hood, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No. 167 to DRP-63
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cc w/encls: See next page

Nine Mile Point Nuclear Station
Unit No. 1

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. DRP-63

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated May 15, 1998, as supplemented by letters dated September 25, October 13, December 9 (two letters), 1998; January 11, April 1, and April 22, 1999, 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 1;

   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. DRP-63 is hereby amended to read as follows:

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(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 167 are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance to be implemented before spent fuel is stored within the new high-density spent fuel rack modules authorized for installation and use by this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Section Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 17, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 167

TO FACILITY OPERATING LICENSE NO. DRP-63

DOCKET NO. 50-220

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by Amendment number and contains vertical lines indicating the areas of change.

Remove	Insert
346	346

5.5 Storage of Unirradiated and Spent Fuel

Unirradiated fuel assemblies will normally be stored in critically safe new fuel storage racks in the reactor building storage vault. Even when flooded with water, the resultant k_{eff} is less than 0.95. Fresh fuel may also be stored in shipping containers. The unirradiated fuel storage vault is designed and shall be maintained with a storage capacity limited to no more than 200 fuel assemblies.

1066 spent fuel assemblies with up to 15.6 grams (3.0 weight percent) of Uranium-235 per axial centimeters of assembly can be stored in non-poison flux trap racks in the north half of the spent fuel pool. 1710 spent fuel assemblies with up to 18.13 grams (3.75 weight percent) of Uranium-235 per axial centimeters of assembly can be stored in Boraflex racks in the south half of the pool. These racks have been designed to maintain a k_{eff} less than 0.95 under conditions of optimum water moderation. The north and south half of the pool are analyzed to store 1840 and 2246 fuel assemblies, respectively, using racks containing the neutron absorbe material Boral. The Boral racks will maintain a k_{eff} of less than 0.95 under abnormal and accident conditions. The spent fuel stored the Boral racks must have a peak lattice enrichment of 4.6 % or less and the k-inf in the standard cold core geometry must be less than or equal to 1.31.

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 11 percent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.167 TO FACILITY OPERATING LICENSE NO. DRP-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated May 15, 1998, as supplemented by letters dated September 25, October 13, December 9 (two letters), 1998; January 11, April 1, and April 22, 1999, Niagara Mohawk Power Corporation (NMPC or the licensee), proposed a license amendment to change the Technical Specifications (TSs) for Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The proposed changes would change TS 5.5, "Storage of Unirradiated and Spent Fuel," to reflect a planned modification to increase the number of fuel assemblies that can be stored in the NMP1 spent fuel pool (SFP) from 2776 (i.e., 1066 in the northern half of the pool and 1710 in the southern half of the pool) to 4086. The changes would also delete an inappropriate reference within TS 5.5 to 10 CFR 70.55 for calculational methods approved by the Commission involving special arrays.

By letters dated September 25, October 13, and December 9, 1998 (two letters); January 11, April 1, and April 22, 1999, NMPC provided additional information in support of the application for amendment. The additional information does not affect the Commission's finding of no significant hazards consideration that was issued in a Federal Register notice (63 FR 64973, November 24, 1998). Similarly, the additional information provided by NMPC does not affect the Commission's environmental assessment and finding of no significant effect upon the quality of the human environment that was issued in a Federal Register notice (64 FR 18059, April 13, 1999).

2.0 BACKGROUND

The NMP1 pool is presently licensed to store up to 2776 fuel assemblies. The northern half of the pool contains 8 racks providing 1066 storage cells of a non-poison flux trap design. The southern half contains 8 racks providing 1710 storage cells that use Boraflex as a neutron absorber.

An increase in spent fuel storage capacity is needed at NMP1 to reestablish full core off-load capability. Loss of that capability occurred as a result of the 1999 refueling outage (RFO15). Thus, after RFO15, NMPC proposes to replace the 8 non-poison rack modules in the northern half of the NMP1 pool with new poison rack modules providing 1840 storage locations. Ultimately, additional capacity will be added to accommodate future refueling outages. Thus, as further capacity increase is warranted by the increasing fuel inventory in the pool, NMPC will increase the capacity of the southern half of the pool to provide a total pool capacity for 4086

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spent fuel assemblies. The capacity of 4086 storage locations is sufficient to extend full core off-load capability to at least the expiration date of the plant operating license, August 22, 2009.

The proposed new high density storage racks were designed by Holtec International (Holtec) and consist of an egg-crate structure of individual cells with 5.9-inch inside square dimension, each of which accommodates a single boiling water reactor (BWR) fuel assembly, with fixed neutron absorber material, known by the trademarked name "Boral," positioned between the fuel assembly storage cells. The nominal center-to-center spacing between stored fuel assemblies is reduced to 6.060 inches. The racks are designed to accommodate standard 8x8 and 9x9 General Electric Company (GE) fuel assemblies.

3.0 EVALUATION

NMPC proposes to change the second paragraph of TS 5.5 to reflect design features for both the existing spent fuel storage design and the proposed modification. Specifically, the paragraph would be changed to state:

> 1066 spent fuel assemblies with up to 15.6 grams (3.0 weight percent) of Uranium-235 per axial centimeters of assembly can be stored in non-poison flux trap racks in the north half of the spent fuel pool. 1710 spent fuel assemblies with up to 18.13 grams (3.75 weight percent) of Uranium-235 per axial centimeters of assembly can be stored in Boraflex racks in the south half of the pool. These racks have been designed to maintain a Keff less than 0.95 under conditions of optimum water moderation. The north and south half of the pool are analyzed to store 1840 and 2246 fuel assemblies, respectively, using racks containing the neutron absorber material Boral. The Boral racks will maintain a Keff of less than 0.95 under abnormal and accident conditions. The spent fuel stored in the Boral racks must have a peak lattice enrichment of 4.6% or less and the K-inf in the standard cold core geometry must be less than or equal to 1.31.

NMPC also proposes to delete the third paragraph of TS 5.5 which states: "Calculations for k_{eff} values have been based on methods approved by the Nuclear Regulatory Commission covering special arrays (10CFR70.55)."

The NRC staff's evaluation of these proposed TS changes are presented in the following sections. The NRC staff's evaluation also includes the design changes and installation activities authorized by the proposed amendment.

3.1 Criticality Evaluation

Appendix A to 10 CFR Part 50, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," states that "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

The analysis of the reactivity effects of fuel storage in the NMP1 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 NMP1 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. The NRC staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the NMP1 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) The racks contain the most reactive fuel authorized to be stored, without any control rods or any uncontained burnable poison, and with the fuel at the burnup corresponding to the highest reactivity during its burnup history.

(2) The pool water is unborated and at the temperature yielding the highest reactivity (4 °C) over the expected range of water temperatures.

(3) An infinite array (no neutron leakage) of storage cells is assumed, except for certain accident assessments.

(4) Neutron absorption in minor structural members is neglected (i.e., spacer grids are analytically replaced by water).

The NRC staff concludes that appropriately conservative assumptions were made.

The design basis fuel assembly used for the criticality analyses is a standard array of 8x8 fuel rods containing UO_2 clad in Zircaloy. An initial uniform U-235 enrichment of 4.0 weight percent (w/o) was assumed, as well as 2% gadolinia (Gd_2O_3) burnable poison in 8 fuel rods. The analysis was performed at the maximum reactivity over burnup, which was found to occur at approximately 8 MWD/KgU. Calculations were also made for other fuel assembly designs previously or currently used in the NMP1 reactor. These additional fuel types are the GE 7x7, GE 8x8 (with 1 water rod), GE 8x8R (with 2 water rods), 8x8 GE8B (with 4 water rods), and the 9x9 GE11 designs.

For the nominal storage cell design, a calculational bias and uncertainty, as well as uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, eccentric positioning, zirconium flow channel bulging, 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 reactivity uncertainty in the depletion calculation, and a 0.01 Δk allowance for possible differences between fuel vendor calculations and those reported here, were included. The final maximum calculated CASMO-3 reactivity resulted in a k-effective (k_{eff}) of 0.935 when combined with all known uncertainties. This meets the NRC staff's criterion of a k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level. Therefore, it is acceptable.

Calculations were also made for storage rack reactivities of fuel enriched to 4.25 and 4.6 w/o U-235, including the GE-11 (9x9 array) fuel enriched to 4.6 w/o U-235, as a function of the fuel assembly k-infinity (k_{inf}) in the standard NMP1 core geometry at 20 °C, defined as an infinite array of fuel assemblies on a 6-inch lattice spacing without any control absorber or voids. The results indicate that a k_{inf} of 1.31 for both the 8x8 and 9x9 fuel designs in the standard core geometry results in a rack reactivity less than 0.95, including all appropriate 95/95 uncertainties, for enrichments up to 4.6 w/o U-235. The 4.6 w/o GE-11 fuel was found to be lower in reactivity than the 4.6 w/o 8x8 fuel type.

Based on these results, a BWR fuel assembly appropriate for use in the NMP1 reactor is acceptable for storage in the NMP1 storage racks if it has a peak lattice enrichment of 4.6 w/o U-235 and if its k_{inf} in the standard NMP1 core geometry, calculated at the maximum over burnup, is less than or equal to 1.31. These requirements are incorporated into the proposed changes to NMP1 TS 5.5. NMPC has also shown that any fuel with an average U-235 enrichment of 3.1 w/o or less is acceptable regardless of the gadolinium content or the k_{inf} in the standard core geometry.

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 or dropping an assembly on top 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.

NMP1 TS 5.5 currently states, in part, that "Calculations for k_{eff} values have been based on methods approved by the Nuclear Regulatory Commission covering special arrays (10CFR70.55)." The specified reference is inappropriate because 10 CFR 70.55 addresses inspections for special nuclear material, not calculational methods. The existing TS statement does not address a required design feature of the facility, which is the purpose of TS Section 5.0. The statement also does not represent any Commission requirement. Therefore, the NRC staff concludes the existing statement is inappropriate and should be deleted.

The following TS changes to TS 5.5 have been proposed as a result of the requested spent fuel pool reracking. The NRC staff finds these changes acceptable.

(1) The number of fuel assemblies which can be stored in the spent fuel pool when all the new Boral racks are installed has been increased from 2776 to 4086.

(2) The spent fuel stored in the Boral racks must have a peak lattice enrichment of 4.6 w/o U-235 or less and the k_{inf} in the standard cold core geometry must be less than or equal to 1.31.

(3) The inappropriate statement involving approved calculational methods covering special arrays, including its reference to 10CFR70.55, is deleted.

Based on the review described above, the NRC staff finds the criticality aspects of the proposed modifications to the NMP1 spent fuel pool storage racks are acceptable and meet the requirements of Appendix A to 10 CFR Part 50,General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

3.2 Spent Fuel Pool Cooling Evaluation

The SFP cooling system (SFPCS) at NMP1 is a Seismic Category 1 system consisting of two cooling trains, each primarily equipped with one pump, one filter and one heat exchanger. The SFPCS is designed with both cooling trains operable and only one cooling train is required to be operating to maintain the SFP water temperature at or below 140 °F during normal (planned) refueling outages (i.e., during a normal (planned) refueling outage at NMP1, an entire core is offloaded). Heat is removed from the SFPCS heat exchangers by the reactor building closed loop cooling system (RBCLCS). The RBCLCS water temperature is maintained between 40 °F and 95 °F depending upon the water temperature of Lake Ontario (the ultimate heat sink).

As a result of the increase in SFAs to be stored in the SFP, the decay heat generated in the SFP will increase. To maintain the SFP water at or below the temperature limit of 140 °F, SFAs must be held in the reactor for a minimum period of time after shutdown before being transferred to the SFP. In any event, based upon radiological exposure requirements, SFAs may not be off-loaded from the reactor prior to a minimum shutdown time of 72 hours. Since the heat removal capability of the SFPCS is a function of RBCLCS water temperature, NMPC performed analyses to determine the reactor shutdown time required before discharging SFAs from the reactor in order to maintain the SFP water temperature at or below 140 °F with RBCLCS water temperatures at 40 °F, 60 °F, 80 °F, and 95 °F. In the analyses, one SFPCS train is assumed to be operating, with both SFPCS trains operable. The following summarizes the results of these analyses:

RBCLCS Water Temp. (°F)	Reactor Shutdown Time Required (hours)	Coincident Time[1] After Reactor Shutdown (hours)	Coincident Net Heat Load (Mbtu/hr)	Time-to-Boil (hours)	Max. Boil-off Rate (gpm)
40	72[2]	177	20.72	8.97	43.72
60	141	250	18.39	8.87	39.05
80	458	573	13.80	11.39	29.41
95	1008	1129	10.35	15.70	22.09

As indicated in the above table, maintaining the SFP temperature limit of 140 °F is based on two primary parameters. The first is the RBCLCS water temperature which, in turn, is a function of the water temperature of Lake Ontario. The second is the SFAs in reactor decay time following reactor shutdown. Therefore, NMPC established the following constraints which are applicable to all full-core discharge operations:

---

1 The time after reactor shutdown at which the SFP water reaches its temperature limit of 140 °F.

2 The calculated peak SFP temperature for this case is 130.1 °F.

1. With both SFPCS trains operable and only one train operating for actual end-of-cycle refueling practice, the SFP water shall be maintained at or below the SFP water temperature limit of 140 °F.

2. In view of the fact that the temperature of the RBCLCS water to the SFPCS heat exchangers can be varied within a wide range (40 °F to 95 °F maximum) depending on the month of the refueling outage and coincident heat load, cycle specific evaluations shall be performed for those times when the temperature of the RBCLCS water to the SFPCS heat exchangers varies from the temperatures as presented in the above table. These cycle specific evaluations will ensure that with specific RBCLCS water temperature and the decay heat generated from the SFAs in the SFP, the SFP water temperature will be maintained at or below 140 °F with one SFPCS train operating. The need for a different hold time, if required, will be addressed as part of the cycle-specific evaluations.

3. The evaluation shall be based upon:

    a. Design basis values of SFP water and RBCLC water flow rates to the SFPCS heat exchangers.

    b. The decay heat load calculation performed in accordance with the provisions of "USNRC Branch Technical Position ASB9-2, " Residual Decay Energy for Light Water Reactors for Long Term Cooling", Rev. 2, July 1981.

    c. The rate of SFAs transferred to the SFP is limited to a maximum of 6 SFAs per hour. However, a more expedited offload may be performed if the plant conditions exist to maintain the pool water temperature at or below 140 °F with one SFPCS train operating or if higher than assumed flows to spent fuel cooling are available.

NMPC stated that the above constraints to fuel discharge operations are incorporated in the appropriate plant procedures.

Also, the SFP has a water temperature monitor system which alarms in the control room when the SFP water temperature reaches 113 °F during plant operation. NMPC stated that plant procedure NI-ARP-I.I provides direction for the operator to increase SFP cooling flow to maintain the SFP temperature between 75 °F and 105 °F when the high temperature alarm is received. Prior to a refueling outage, the SFP temperature setpoint is administratively lowered to 100 °F. If the SFP temperature reaches 100 °F, precautious actions will be taken in accordance with the guidance described in plant procedure NI-OP-34. This will provide additional assurance that the SFP temperature limit of 140 ^{0}F is not exceeded.

On the basis of the NRC staff's review and NMPC's statements, and with the above cited constraints incorporated into the appropriate plant procedures, the NRC staff finds that none of the SFAs discharge scenarios will result in the SFP exceeding its design temperature of 140 °F.

In the unlikely event of a complete loss of cooling, the SFP water temperature would begin to rise and, in the absence of operator action, would eventually reach the boiling temperature. The calculated minimum time-to-boil is 8.87 hours and the maximum boil-off rate is 43.72 gpm. There is, therefore, ample time for operator action. This calculated maximum boil-off rate is within the available make-up rate of 75 gpm by the condensate transfer system. In addition, as a backup to the make-up water system, water can also be supplied directly to the SFP from Lake Ontario via the fire protection system.

On the basis of its review, the NRC staff finds that in the unlikely event of a complete loss of cooling, NMPC has sufficient time and is capable of aligning make-up water to the pool before boiling begins and that the make-up water will be supplied at a rate which well exceeds the boil-off rate.

Although NMP1 is a pre-SRP (Standard Review Plan) plant, NMPC also performed the two discharge scenarios (partial core discharge and abnormal full core off-load) in accordance with the guidance in SRP 9.1.3, "Spent Fuel Pool Cooling And Cleanup System," to illustrate compliance with the provisions of SRP 9.1.3. The following summarizes the results of these two SRP discharge scenarios with RBCLCS water temperature at 95 °F:

Discharge Scenarios	RBCLCS Water Temp. (°F)	Peak SFP Water Temp. (°F)	Coincident Time After Reactor Shutdown (hours)	Coincident Net Heat Load (Mbtu/hr)	Time-to-Boil (hours)	Max. Boil-off Rate (gpm)
Partial Core	95	119.8	208	5.70	37.39	11.61
Abnormal Full Core-offloaded	95	138.3[3]	266	19.92	8.37	42.35

As indicated in the above table, for the case of the partial core discharge scenario with one train of SFPCS operating, the calculated peak SFP temperature is 119.8 °F, which is below the SRP temperature limit of 140 °F for SFP. For a complete loss of SFP cooling, the minimum time-to-boil is 37.39 hours and the maximum boil-off rate is 11.61 gpm, which is well below the 75 gpm available make-up rate.

For the case of abnormal full-core discharge scenario with both trains of SFPCS operating, the calculated peak SFP temperature is 138.3 °F, which satisfies the SRP guidance of no boiling. For a complete loss of SFP cooling, the minimum time-to-boil is 8.37 hours, and the maximum boil-off rate is 42.35 gpm, which is well below the 75 gpm available make-up rate.

On the basis of the NRC staff's review of NMPC's analyses, and with the above cited constraints for all SFA discharge scenarios with SFP cooling at various RBCLCS water temperatures incorporated into the plant operating procedures, the NRC staff concludes that NMPC's proposed plan to install additional racks in the SFP at NMP1 following the 1999 refueling outage to allow, in two campaigns, an increase in the spent fuel storage capacity from

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3 With both SFP cooling trains operating.

2776 to 4086 SFAs is acceptable. The NMP1 final safety analysis report (FSAR) will be updated to reflect the above information regarding SFP cooling.

3.3 Materials Compatibility Evaluation

3.3.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel racks include: ASME SA240-304L for all sheet metal stock and internally threaded support legs, ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100 °F) for externally threaded support spindle, and ASME Type SFA 5.9 R308L for weld material.

The NRC staff finds that these materials used in the Holtec racks have a favorable history of in-pool usage. They are compatible with the spent fuel assemblies and the spent fuel environment. Therefore, they are acceptable for use in this application.

3.3.2 Poison Material

The Holtec racks employ Boral™ as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the SFP environments where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum, typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. If not allowed to escape, hydrogen gas, a product of the corrosion process, could cause swelling in the rack panels resulting in deformation of the storage cells. To prevent the storage cell deformation, the Boral panels are held in place by a stainless steel sheathing that is spot-welded to the cell walls. The sheathing is not seal-welded and, thus, gas released from the Boral panel may escape from the sheathing. Therefore, it is not possible for pressure to build up in the sheathing. The neutron absorbing capability of Boral is not affected by this corrosion process.

Boral has a long history of applications in nuclear plants and its use has indicated that the material exhibits a high degree of stability to radiation. Based upon accelerated test programs, Boral is considered a satisfactory material for reactivity control in spent fuel storage racks and is fully expected to fulfill its design function over the lifetime of the racks. Nevertheless, NMPC has established a surveillance program. This surveillance program includes the use of coupons to obtain physical and chemical properties from which the stability and integrity of the Boral in the storage cells may be inferred.

3.3.3 Conclusions

Based on its evaluation, the NRC staff finds that the materials utilized in the fabrication of the spent fuel racks manufactured by Holtec International are of proven durability and are compatible with the SFP environment at NMP1. The type of degradation exhibited by the racks does not affect their neutron absorbing capability and the rack design allows for the venting of the corrosion product gas, hydrogen, to prevent deformation of the storage cells. In addition, the NMPC has established a Boral integrity surveillance program to verify the continued, satisfactory performance of Boral throughout the life of the spent fuel racks. Based on this evaluation, the NRC staff concludes that the materials used in the new spent fuel racks are acceptable and will perform their structural and neutron absorbing functions satisfactorily in the NMP1 pool environment.

3.4 Structural Evaluation

The primary purpose of this review is to evaluate the structural integrity and functionality of the racks, the stored fuel assemblies, and the SFP structure subject to the effects of the postulated loads (Appendix D of Standard Review Plan (SRP) Section 3.8.4) and fuel handling accidents.

3.4.1 Storage Rack Integrity

The total storage capacity of 4086 fuel assemblies will be accomplished by adding 16 fuel storage racks, which are seismic Category I equipment and, thus, are required to remain functional during and after a safe shutdown earthquake (SSE). The first 8 racks will be placed in the SFP following NMP1's 1999 refueling outage (Campaign I) and the remaining 8 racks will be placed at a later time (Campaign II). NMPC, with its contractor Holtec International, have performed structural analyses of the racks in support of the requested license amendment.

The computer program DYNARACK was used for dynamic analysis to demonstrate the structural adequacy of the NMP1 spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor or walls of the SFP. 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 the three dimensional (3-D) 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 (SR) model analysis and the 3-D whole pool multi-rack (MR) model analysis. In the 3-D SR analysis, the rack was considered to be fully loaded, half loaded, and almost empty with two different coefficients of friction ($\mu=0.2$, and 0.8) between the rack pedestal and the pool floor to investigate the stability of the rack with respect to overturning. In the 3-D MR analysis, 16 free-standing racks were considered fully loaded and partially loaded, with a random coefficient of friction (mean is about 0.5) to investigate the fluid-structure interaction effects between the racks and the pool walls, as well as those among the racks, and to identify the worst case response for rack movement and for rack member stresses.

The seismic analyses were performed using the direct integration, time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) was generated from the design response spectra defined in the NMP1 FSAR. NMPC demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra, as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in SRP Section 3.7.1.

A total of 30 3-D SR and MR analyses were performed. The racks were subjected to the service, upset, and faulted loading conditions (Level A, B and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 1.37 inches, indicating that there is adequate safety margin against overturning of the racks. In addition, 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 (the ASME Code), Section III, Subsection NF. The results show that all induced stresses under the SSE loading condition are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the rack design is adequate.

NMPC also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions. NMPC demonstrated that all the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate.

On the basis of: (1) NMPC's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowable values provided in the ASME Code, and (3) NMPC's overall structural integrity conclusions supported by both SR and MR analyses, the NRC staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions. Therefore, the proposed rack modules are acceptable.

3.4.2 Spent Fuel Storage Pool Integrity

NMPC analyzed the SFP using the finite element computer program, ANSYS, to demonstrate the adequacy of the structures under fully-loaded fuel rack conditions (i.e., with all storage locations occupied by fuel assemblies). The fully-loaded structures were subjected to the load combinations specified in the NMP1 FSAR.

Tables 8.5.2, 8.5.3 and 8.5.4 in NMPC's letter dated May 15, 1998, show the predicted factors of safety varying from 1.32 to 4.83 for axial force, shear force, and bending moments of the concrete walls and slab. In view of the calculated factors of safety, the NRC staff concludes that NMPC's structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading and SSE loading conditions. Thus, the structural aspects of the fuel storage pool design are acceptable.

3.4.3 Fuel Handling Accident Integrity

The following two refueling accident cases were evaluated by NMPC: (1) the dropping of a fuel assembly and its handling tool, which impacts the baseplate (i.e., the "deep drop" scenario) and (2) the dropping of a fuel assembly and its handling tool, which impacts the top of a rack (i.e., the "shallow drop" scenario).

The results of the analysis of the deep drop scenario show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area. Therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The results of the analysis of the shallow drop scenario show that damage would be restricted to a depth of 7.3 inches below the top of the rack, which is above the active fuel region. The NRC staff reviewed NMPC's analysis results in NMPC's letter dated May 15, 1998. The NRC staff finds that NMPC's structural integrity conclusions are appropriately supported by the parametric studies, and the NRC staff, therefore, concurs with NMPC's findings.

3.4.4 Conclusion

Based on its review and evaluation of NMPC's submittal dated May 15, 1998, and additional information regarding structural evaluations provided by NMPC in letters dated December 9, 1998, and April 22, 1999, the NRC staff concludes that NMPC's structural analysis and design of the spent fuel rack modules and the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with the current licensing basis set forth in the FSAR and applicable provisions of the SRP, and are, therefore, acceptable.

3.5 Occupational Radiation Exposure

The NRC staff has reviewed NMPC's plan for the modification of the NMP1 SFP storage racks with respect to occupational radiation exposure. As previously noted, for this modification NMPC plans to ultimately install a total of 16 new fuel rack modules in the SFP and cask storage pit. A number of nuclear power facilities have performed similar operations in the past. With the benefit of the lessons learned from these previous operations, NMPC estimates that the proposed fuel rack installation can be implemented while maintaining occupational radiation exposure between 6 and 12 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of as-low-as-is-reasonably-achievable (ALARA) principles. NMPC's Radiation Protection department will prepare Radiation Work Permits for the various jobs associated with the reracking operation. Each member of the project team will receive radiation protection training on the reracking operation. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment consisting, at a minimum, of thermoluminescent dosimeters (TLDs) and self-reading dosimeters.

NMPC may also use divers for the removal of the existing SFP rack modules and installation of the replacement high-density racks. These divers may also be needed to remove certain underwater appurtenances in the SFP. Each diver will be equipped with whole body and extremity dosimetry with remote, above surface, readouts that will be continuously monitored by NMPC's Radiation Protection personnel. Divers will also be equipped with underwater survey instrumentation with remote readout capabilities. NMPC will utilize underwater cameras to permit remote monitoring of the diver's location at all times. Divers will also be in continuous communication with the Radiation Protection personnel. NMPC will conduct radiation surveys of the diving area before each diving operation and following the movement of any irradiated hardware in the SFP. NMPC will use either visual or physical barriers to ensure that divers maintain a safe distance from spent fuel assemblies or other high radiation sources in the SFP. NMPC will also use a safety line attached to the diver and manned by a dive tender at all times to maintain positive diver control.

NMPC will use a pressure washer or other acceptable cleaning mechanism to decontaminate the existing SFP rack modules (as well as any interferences or SFP hardware that must be removed from the SFP) prior to removal from the SFP. All items removed from the SFP will be closely monitored for hot particles. Once the SFP racks and other hardware are removed from the SFP, they will be rinsed with demineralized water, packaged, and placed in a special shipping container approved by the U.S. Department of Transportation. The removed SFP racks will then be shipped offsite to a licensed processing/disposal facility. NMPC does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase due to the expanded SFP storage capacity. However, NMPC will operate continuous air monitors in areas where there is a potential for significant airborne activity during the fuel reracking operation. In addition, the Reactor Ventilation Radiation Monitor will be used to monitor airborne activity.

NMPC will monitor and control personnel traffic and equipment movement in the SFP area to minimize contamination and to assure that radiological exposures are maintained ALARA. NMPC plans to use an underwater vacuum cleaner system to remove crud and debris from the bottom of the SFP following removal of the old SFP rack modules. This vacuum system will also be used to capture metal filings generated by any cutting performed in the SFP. NMPC will use the existing SFP filtration system during fuel rack installation to maintain water clarity in the SFP.

The storage of additional spent fuel assemblies in the SFP will result in negligible increases in the dose rates on the refueling floor and in adjacent areas accessible to the SFP. Maximum dose rates outside the concrete wall of the SFP will be less than 5 mr/hr, and dose rates below the concrete floor of the SFP will be less than 0.8 mr/hr.

On the basis of its review of the NMPC's plans and preparations to control radiological exposures, the NRC staff concludes that the NMP1 SFP rack modification can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The NRC staff finds the projected dose for the project of 6 to 12 person-rem to be within the range of doses for similar SFP modifications at other plants. The projected dose is, therefore, acceptable.

3.6 <u>Solid Radioactive Waste Evaluation</u>

Spent resins are generated by the processing of SFP water through the SFP purification system. NMPC predicts that the resin changeout frequency of the SFP purification system may be increased temporarily during the reracking operation. To maintain the SFP water as clean as possible, and thereby minimize the generation of spent resins, NMPC will vacuum the floor of the SFP to remove any radioactive crud, sediment, and other debris before the new fuel rack modules are installed. Filters from this underwater vacuum will be a source of solid radioactive waste (radwaste). Additional solid radwaste will consist of the old SFP rack modules themselves, as well as any interferences or SFP hardware that may have to be removed from the SFP to permit installation of the new spent fuel rack modules. Overall, however, NMPC does not expect that increasing the storage capacity of the SFP will result in a significant change in the generation of solid radwaste at NMP1.

3.7 Fuel Handling Accident Dose Evaluation

In Section XV.C.3 of the NMP1 FSAR, NMPC evaluated the possible radiological consequences of a fuel handling accident (FHA). The proposed reracking of the NMP1 SFP will not affect any of the assumptions or inputs used in evaluating the dose consequences of the FHA.

The NRC staff reviewed NMPC's analysis and performed confirmatory calculations to check the acceptability of NMPC's doses. For its calculations, the NRC staff used the assumptions of RG 1.25, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." For a FHA in the Reactor Building, the NRC staff assumed that the cladding of 125 fuel rods (the equivalent of two full 8x8 fuel assemblies) would be ruptured if a fuel assembly were dropped during handling. The damaged fuel rods are assumed to contain freshly off-loaded fuel with a minimum of 24 hours of decay. The parameters that the NRC staff used in its assessment are presented in Table 3.7-1 of this safety evaluation.

The staff's calculations confirmed that the thyroid doses at the Exclusion Area Boundary (EAB), Low-Population Zone (LPZ), and Control Room from a fuel handling accident meet the acceptance criteria and that NMPC's calculations are acceptable. The results of the NRC staff's calculations are presented in Table 3.7-2 of this safety evaluation. For a FHA, the NRC staff calculated a dose of 0.68 rem to the thyroid at the EAB and 0.41 rem to the thyroid at the LPZ. The acceptance criterion at the EAB and LPZ for these accidents is 75 rem for the thyroid dose (25 percent of 10 CFR Part 100 guidelines of 300 rem, as stated in SRP Section 15.7.4). The NRC staff calculated a dose to the control room operator of 4.5 rem to the thyroid. The acceptance criterion for the control room operator is 30 rem to the thyroid (SRP Section 6.4). Therefore, the NRC staff finds the proposed reracking operation at NMP1 to be acceptable with respect to potential radiological consequences as a result of a hypothetical fuel handling accident.

Table 3.7-1

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT AT NINE MILE POINT UNIT 1

Parameters	
Power Level, Mwt	1887
Number of Fuel Rods Damaged (2 assemblies)	125
Total Number of Rods in Core	32,984
Shutdown Time, hours	24
Power Peaking Factor	1.5
Fission-Product Release Fractions (%)*	
Iodine (corrected for extended burnup)	12
Noble Gases	30
Pool Decontamination Factors*	
Iodine	100
Noble Gases	1
Iodine Forms (%)*	
Elemental	75
Organic	25
Filter Efficiencies for Control Room (%)	
Elemental	0
Organic	0
Atmospheric Dispersion Factors, X/Q (sec/m^3)	
Exclusion Area Boundary (0-2 hours)**	4.7×10^{-5}
Low Population Zone (0-8 hours)**	2.8×10^{-5}
Control Room (0-8 hours)	3.12×10^{-4}
Dose Conversion Factors per ICRP 30	

* Regulatory Guide 1.25
** NRC staff calculated

TABLE 3.7-2

THYROID DOSES FROM FUEL HANDLING ACCIDENT AT NMP1 (VALUES CALCULATED BY NRC STAFF)

	DOSE (REM)
	FUEL HANDLING ACCIDENT
EAB*	0.68
LPZ*	0.41
Control Room**	4.52

*Acceptance Criterion = 75 rem thyroid
**Acceptance Criterion = 30 rem thyroid

3.8 Heavy Loads Handling Evaluation

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides guidelines and recommendations for licensees to assure safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over spent fuel assemblies, over the core, and over safety-related equipment. The NUREG defines a heavy load as any load carried in a given area during the operation of the plant that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool.

Phase I of NUREG-0612 provide guidelines for reducing the likelihood of dropping heavy loads and limiting the resulting potential consequences of a drop. The guidelines are focused on establishing safe load paths, procedures for load handling operations, training of crane operators, the design of lifting devices and the design, testing, inspection, and maintenance of cranes. Phase II of NUREG-0612 provides guidelines for mitigating the consequences of dropped loads, including the use of a single failure proof crane, use of electrical interlocks and mechanical stops to restrict crane travel, and performance of load drop and consequence analyses to assess the impact of dropped loads on plant safety. Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, in GL 85-11, licensees were encouraged to implement actions they perceive to be appropriate to maintain safety. The NRC staff reasserted this position in NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment." In NRC Bulletin 96-02, the NRC staff also alerted licensees to the importance of complying with existing regulatory guidelines on the control and handling of heavy loads. Licensees were also reminded of their responsibilities for providing adequate protection of public health and safety when handling heavy loads during plant operation.

The NMP1 spent fuel storage pool is divided into two halves, currently with the capacity to store 1066 and 1710 spent fuel assemblies for a total of 2776 storage cells. NMPC has requested a license amendment to change TS 5.5 to reflect a proposed increase the storage capacity of the SFP from 2776 to 4086 storage cells.

The NMP1 SFP has undergone four previous phases of rack installation--in 1978, 1982, 1994, and 1996. In the proposed reracking, NMPC plans to install high density stainless steel racks in two phases. In the first phase, the non-poison racks in the northern half of the SFP will be replaced with poison racks after completion of the 1999 refueling outage. In the second phase, NMPC will rerack the southern half of the SFP at a later date as additional storage capacity is needed. NMPC will install 1840 and 2246 storage cells in the northern and southern halves of the SFP, respectively.

In this and previous rerackings, NMPC addressed issues pertaining to NUREG-0612. Specifically, NMPC addressed activities involving the handling and control of heavy loads, including movement of spent fuel assemblies, removal and installation of spent fuel storage racks, the design and use of the Reactor Building hoisting system, establishment of safe load paths, the use of procedures, crane operator training, and postulated load drop accident analyses and consequences over the SPF and safety-related equipment.

3.8.1 Hoisting System

NMPC plans to use the Reactor Building's 125-ton overhead crane to handle heavy loads during the rerack operation. As stated by NMPC in its response to NRC Bulletin 96-02, dated May 13, 1996, the reliability of the Reactor Building crane was upgraded to single-failure-proof in accordance with criteria provided in Section 5.1.6 of NUREG-0612. Therefore, the crane has an increase in the factor of safety to 10 to 1. In addition, the main hoist of the crane is equipped with a redundant hoisting system to prevent dropping the rack in the event of failure of a single hoist component. The crane is designed in accordance with the requirements of CMAA No. 70, "Specifications for Electric Overhead Traveling Cranes, " and ANSI B30.2-1976, "Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)." The lifting capacity of the crane is 125 tons in the main hoist, and 25 tons and 1000 pounds in two auxiliary hoists.

A remotely controlled lifting rig will be used with the Reactor Building overhead crane to handle the spent fuel racks. NMPC states that the lifting rig is specifically designed to handle the spent fuel racks and is physically identical to the rigs used by a number of licensees. The lifting rig is designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6 (1978), "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials." It consists of redundant lift rods and connecting lift eyes. The lift rods are independently loaded and configured such that failure of a single rod will not result in uncontrolled lowering of the rack. NMPC states that both the stress design and the load testing of the lifting rig satisfy guidelines in Section 5.1.6(1)(b) of NUREG-0612 and ANSI N14.6 (1978), respectively. Accordingly, the lift rods are designed as follows: (1) with at least twice the normal stress design factor (a safety factor of 10 to 1); (2) load tested to three times the maximum weight to be lifted plus the weight of the lift rig, (3) the total weight to be lifted is sustained for approximately 1 hour; and (4) after load testing, the integrity of the critical joints and welds are examined using a liquid penetrant.

The NRC staff believes that the design and testing of the lifting rig and other lifting devices, and the limit on lifts made with the overhead crane, will enable NMPC to safely handle heavy loads with little to no risks to workers or equipment during the rerack operation.

3.8.2 Analysis of Postulated Load Drop Accidents

NMPC evaluated the consequences of dropping a spent fuel rack module from the top of the water surface onto the SFP floor. Based on this evaluation, NMPC concluded that: (1) the SFP would not experience any damage that would result in water leakage and uncovering the fuel, and (2) the potential for a rack drop to impact fuel is unlikely because the NMPC plans to shuffle any fuel in the load path to racks outside the load path before any load movement. In a previous response to NRC Bulletin 96-02, NMPC concluded that, in accordance with NUREG-0612, a load drop assessment for a potential rack drop onto safe-shutdown equipment was not needed due to the single-failure-proof reliability of the handling system.

NUREG-0612 recommended that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel, and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. NMPC stated that it will implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. NMPC plans to provide comprehensive training to the rerack installation crew, use redundantly designed lifting rigs, and perform inspection and maintenance checks on the cranes and lifting devices before the rerack operation. In addition, NMPC will restrict travel of the racks from over fuel in the SFP and safety-related equipment, limit the lift heights of the racks to 6 inches above the SFP floor prior to any vertical lifts from the SFP, and limit the lifts to less than 50 percent of the crane's capacity. The NRC staff agrees with NMPC that its plans to use administrative procedures and controls focused on, but not limited to, these areas are in accordance with NUREG-0612 and, therefore, would enhance the safety of the rerack operation.

3.8.3. Conclusion

On the basis of the preceding discussion, the NRC staff finds that NMPC's considerations for the movement of heavy loads to support proposed changes to accomplish the SFP reracking in accordance with the proposed license amendment are acceptable. NMPC's use of the Reactor Building overhead crane and the lifting rig, and NMPC's administrative controls and procedures are in accordance with the guidelines in NUREG-0612 and ANSI N14.6. The increased reliability of the crane coupled with the design, testing and inspection of the lifting rig, will enable NMPC to safely handle the racks and other heavy loads during the rerack operation. The administrative controls and procedures to improve the handling and control of the racks further enhance NMPC's capability to reduce the potential for a load drop. Therefore, the NRC staff concludes that the changes to the SFP capacity and the associated TS changes for storing spent fuel assemblies are acceptable.

4.0 <u>STATE CONSULTATION</u>

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

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has previously been prepared and published in the Federal Register on April 13, 1999 (64 FR 18059).

Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect upon the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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