February 1, 1984

Docket No. 50-220

Mr. G. K. Rhode
Senior Vice President
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Rhode:

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications and is in response to your letters dated March 22, 1978, December 20 and 21, 1978, February 26, 1981, June 24, August 5, October 5, October 26, November 18 and December 21, 1983, and January 3, 1984.

This amendment revises the Technical Specifications to allow an increase in the spent fuel storage capacity at the Station from 1984 to a maximum of 2776 assemblies by use of neutron absorbing spent fuel storage racks. Further additional restrictions are imposed regarding storage of spent fuel based on enrichment.

A copy of the Safety Evaluation, Environmental Impact Appraisal and Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Original signed by/

Robert A. Hermann, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 54 to License No. DPR-63
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice/Negative Declaration

cc w/enclosures:
See next page

DISTRIBUTION	Docket File	RHermann	WJones
NRC PDR	OELD	DBrinkman	Extra - 5
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HDenton	ELJordan	RFerguson	TBarnhart(4)
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*Please see previous concurrence page.

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01/20/84	01/20/84	01/20/84	01/25/84	01/26/84	01/30/84	02/1/84

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Mr. G. K. Rhode
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station, Unit No. 1

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. DPR-63

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated March 22, 1978, supplemented December 20 and 21, 1978, February 26, 1981, June 24, August 5, October 5, October 26, November 18 and December 21, 1983 and January 3, 1984, 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. DPR-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 54, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

DB Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 1, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise the Appendix A Technical Specifications by removing and inserting the following pages:

Existing Page	Revised Page
244	244

The revised areas are indicated by marginal lines.

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.

The spent fuel storage facility is designed to maintain fuel in a geometry such that k_{eff} is less than 0.95 under conditions of optimum water moderation. The spent fuel storage facility is designed and shall be maintained with a storage capacity limited to no more than 2776 fuel assemblies. Fuel assemblies stored in the 1066 spent fuel storage locations of the non-poison flux trap design are limited to 15.6 grams (3.0 weight percent) of Uranium-235 per axial centimeters of assembly. Fuel assemblies stored in the 1,710 spent fuel storage positions of the poison type which use Boraflex as the neutron absorber are limited to 18.13 grams (3.75 weight percent) of Uranium-235 per axial centimeters of assembly.

Calculations for k_{eff} values have been based on methods approved by the Nuclear Regulatory Commission covering special arrays (10CFR70.55).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 Introduction

By letters dated March 22, 1978, supplemented by letters of December 20 and 21, 1978, February 26, 1981, June 24, August 5, October 5, October 26, November 18 and December 21, 1983 and January 3, 1984, Niagara Mohawk Power Corporation (the licensee) requested an amendment to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP-1). The request is to authorize increased storage capability in the spent fuel pool (SFP) for the nuclear unit. The proposed modification would increase the SFP storage spaces. This expanded storage capacity will allow the continued operation of the unit through the 1994 refueling outage with full core discharge capability.

The licensee's proposal would increase the SFP storage capacity by replacing the original existing spent fuel storage racks in the south half of the pool with new high density storage racks. The new racks will contain neutron absorber material in separate rectangular containers so that spacing between stored assemblies can be reduced while maintaining adequate criticality margin. The new 1710 spaces are contained in eight high density racks made up of approximately 6 by 12 inches rectangular cross section fuel containers spaced by approximately 1.7 by 12 inches rectangular cross section poison container with two sheets 0.110 inches boroflex poison. The cells making up the module have 7.81-inch center-to-center spacing. The spacing is sufficient to maintain K_{eff} below 0.95. The racks are also designed in such a manner that accidental dropping of a fuel assembly will not cause a geometry that could result in criticality.

The staff evaluation of the safety considerations associated with this proposed action are addressed below. A separate Environmental Impact Appraisal has been prepared for this action.

Notice of Proposed Issuance of Amendment to Facility Operating License No. DPR-63 issued to Niagara Mohawk Power Corporation was published in the Federal Register on November 7, 1978 (43 FR 51883).

2.0 Discussion and Evaluation

2.1 Structural and Mechanical Design Considerations

Description of the Spent Fuel Pool and New Racks

Nine Mile Point Unit 1 is a Mark (Mk.) 1 Boiling Water Reactor (BWR). The plant is founded on rock. The spent fuel pool is located in the reactor building and only serves this unit. As is typical with Mk. 1 BWRs, the pool is located well above the basemat. The top of the pool is at elevation 340.08 ft. (all elevations are from sea level) while the bottom (interior) is at elevation 301.17 ft. The top of the basemat is at elevation 212.0 ft. The inside dimensions of the pool are approximately 37.7 ft. wide by 33.2 ft. long by 38.9 ft. deep. The walls of the pool are 6 ft. thick reinforced concrete and the floor is about 5.7 ft. thick reinforced concrete. The floor is thickened to about 6.7 ft. in the shape of a cross which bisects the pool. The pool is supported by 7 reinforced concrete columns, each 4.5 ft. square and, at one corner, by the 7.0 ft. thick shield wall which surrounds the reactor. The columns are placed at the corners of the pool, at the approximate center of each perimeter wall, and directly in the center of the pool floor.

The pool is lined with a continuous, watertight, 1/4 inch thick stainless stainless steel liner plate. A grid of 3/4 inch thick embedded plates and anchor bolts supports the liner and also provides anchorage for a system of clips which are used to provide lateral restraint for spent fuel racks. A leak-chase channel system is provided in order to detect leaks.

The new racks are stainless steel boxes with individual cells provided for each fuel bundle. The fuel cells are separated by dividers or poison cells. The rack cells are constructed of 0.093 inch thick cold-formed material. Individual fuel and poison cells are welded at the bottom to a heavy base assembly and to each other at the top. Each rack is supported on 4 corner pedestals which are welded to the base of the rack. The cells are fusion spot welded to each other along their height on all sides of each cell.

The 216 cell rack is approximately 92 inches wide by 108 inches long by 178 inches high including the pedestals. The new racks are restrained against gross (over 1/4 inch) horizontal movement by brackets at the walls and by a series of "seismic" beams attached by bolting to the clips mentioned above. The racks are free to move vertically.

Evaluation

Applicable Codes, Standards and Specifications

The racks were designed to conform to the staff's reuqirements as outlined in Appendix D of the USNRC Standard Review Plan (NUREG 0800), Section 3.8.4. As such, the racks were designed to meet the requirements of Section III, Division 2, Subsection NF of the ASME code. The existing pool was designed to the requirements of ACAI 318-63; however, the spent fuel pool structure was evaluated to meet the requirements of ACI 349-76 for this modification.

Accordingly, the codes, standards and specifications used for the design of the racks and the analysis of the pool are acceptable.

Loads and Load Combinations

a. Loads and load combinations for the design of the racks are in accordance with Appendix D to SRP Section 3.8.4 and are acceptable.

b. Loads and load combinations for the analysis for the pool are in accordance with ACI 349-76 and are acceptable.

c. Base seismic input time histories were taken from Unit 2. These records are based on an acceleration of 0.15g and produce response spectra which envelop the Regulatory Guide 1.60 response spectra. Peak broadened (±15%) floor response spectra were developed for the appropriate elevations and time histories were synthesized whose response spectra enveloped the peak broadened response spectra. These synthesized time histories were then used as input in the analysis of the racks. However, since the base acceleration level for Nine Mile Point Unit 1 is 0.11g, the input acceleration levels were multiplied by 0.11/0.15 = 0.733 in the analysis of the racks. These seismic load inputs are acceptable.

d. In addition, loads and load combinations were considered for a fuel-drop accident and for the postulated stuck fuel assembly. These loads were found to be acceptable.

Materials

Materials for the racks are specified to be in conformance to the ASME Code and this is acceptable.

Design and Analysis Procedures

a. Racks

For horizontal directions, a detailed, non-linear time-history analysis of the racks was conducted in order to define seismic loads. Fuel-to-rack interactions, rack-to-pool floor interactions, effects of water mass and friction effects were satisfactorily accounted for. A response spectra approach was used for seismic analysis in the vertical direction. For each direction, components of force from each analysis were combined by the SRSS method. Seismic loads were then combined with other loads, as noted above, for the design/analysis for the rack components and welds. Results were found to be satisfactory.

b. Pool

Impact loads from the racks, as determined above for simultaneous lift-off of all the racks, plus other seismic loads due to the weight of the pool structure and water plus other appplicable thermal and dead load components were combined and applied to a detailed finite element analysis of the pool. The analysis and results were found to be satisfactory.

2.2 Materials Considerations

We have reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water. In addition, our review has included an evaluation of the Boraflex neutron absorber material used in the high density storage locations for environmental stability.

There will be flux trap high density fuel storage racks, poison type high density fuel storage racks and work tables in the Nine Mile Point - 1 spent fuel storage pool for an extended period of time following the modification. The spent fuel pool is filled with demineralized high-purity, high resistivity water. The new high-density spent fuel storage racks are of welded 300 series stainless steel construction with a Boraflex neutron absorber sandwiched between the stainless steel sheets. The neutron absorber is composed of boron carbide powder in a rubber-like silicone polymeric matrix. The flux trap high-density spent fuel storage racks, the work tables, the rack support structure as well as the pool liner are fabricated from 300 series stainless steels.

The inherent high corrosion resistance of stainless steel make it well suited for use in demineralized water at the pool service temperatures. Stainless steel fuel storage racks submerged in water have been in use for 20 years with no deterioration evident. In this environment of oxygen-saturated high purity water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.0×10^{-5} inches in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar galvanic potentials.

The Boraflex poison material is composed of nonconductive materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material.

The space which contains the Boraflex is vented to the pool. Venting will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the stainless steel tube.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long term fuel storage cell surveillance program. Surveillance samples are in the form of removable stainless steel clad Boraflex sheets, which are proto-typical of the fuel storage cell walls. These specimens will be removed and examined periodically.

From our evaluation as discussed above, we find that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the remaining life of the plant. Components in the spent fuel storage pool are constructed of similar alloys and, therefore, have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in water indicate that the Boraflex material will not undergo significant degradation during the expected service life of 40 years.

We further find that the environmental compatibility and stability of the materials used in the spent fuel storage pool are adequate, based on test data and actual service experience in operating reactors.

We have reviewed the surveillance program and find that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensee, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool.

2.3 Installation and Heavy Load Handling Considerations

Prior to beginning the operations required to rerack the spent fuel storage pool, all fuel will be removed from the original storage racks and the work platforms at the south end of the storage pool, and this fuel will be placed in the storage racks at the north end of the pool. Therefore, no heavy load handling operations will be required above stored spent fuel assemblies during the reracking of the storage pool.

In regard to the general load handling procedures to be followed during the reracking of the spent fuel pool, the licensee has indicated the following commitments:

1. Figure 1 of the licensee's November 18, 1983 submittal illustrates the safe load paths that will be followed by heavy loads during reracking of the pool. In lieu of marking the safe load paths on the operating floor, the licensee will utilize a signalman to assist the crane operator in maintaining the load on the safe load path during these operations.

2. Load handling procedures will be utilized which include the following: identification of proper handling equipment, safe load paths, and the required inspections and acceptance criteria before movement of the loads.

3. Prior to moving loads, a lesson guide will be in place and used by the crane operator. This guide meets the intent of ANSI B30.2-1976 as it relates to the training, qualification and conduct of crane operators.

4. The special and general purpose lifting devices utilized in reracking the spent fuel pool meet the requirements of ANSI B14.6 and ANSI B30.9.

5. The crane will be inspected prior to use. The inspection will incorporate the requirements of ANSI B30.2 as it relates to maintenance.

6. The crane used in handling the heavy loads has been designed in accordance with CMAA-70-1981 and ANSI B30.2.

We have reviewed the above commitments in relation to the general load handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". Based on this review, and the fact that heavy loads will not be handled above stored spent fuel during the reracking of the Nine Mile Point Unit 1 spent fuel pool, we conclude that the load handling operations have been adequately addressed and therefore are acceptable.

2.4 Spent Fuel Pool Cooling Considerations

2.4.1 Decay Heat and Spent Fuel Pool Cooling Systems

An evaluation of the decay heat loads identified in the licensee's March 21, 1978 submittal was previously made by the staff. In that evaluation, it was conservatively assumed that the full spent fuel pool expansion would result in 3009 filled storage locations. The June 24, 1983 submittal indicates that the total maximum storage capacity was reduced to 2776. Due to the reduction of the total storage capacity and the more detailed information on the previous and projected discharges given in the November 18, 1983 submittal, the staff recalculated the maximum normal and abnormal heat loads in accordance with the guidance of Standard Review Plan - Section 9.1.3. In both heat load cases, the resulting heat loads have slightly changed from those in our previous safety evaluation. Based on these results, we concur with the heat loads presented in Tables 2.0 and 3.0 of the November 18, 1983 submittal. With the maximum normal heat load assumed, and only one of the two cooling trains in operation, the pool water temperature is calculated to 125 F which is below the 140 F limit recommended in SRP Section 9.1.3. When the maximum abnormal heat load is assumed, and two cooling trains are operating, the maximum pool water temperature is calculated to be 124 F which is below the boiling temperature limit as set forth in the guidance of SRP Section 9.1.3. Therefore, the staff concludes, as in the previous review, that the spent fuel pool cooling system adequately meets the acceptance criteria of SRP Section 9.1.3, and is therefore acceptable.

The above equilibrium temperatures are based on a service water inlet temperature of 90 F. The maximum service water inlet temperature is stated by the licensee to be 95 F. Assuming the maximum service water temperature of 95F, the above pool water temperatures would increase by roughly 5 F. This would not cause the above cited pool temperatures to exceed the limits identified in SRP Section 9.1.3. In addition, the licensee has committed to the following. "Refueling and core offloading operations will not begin until it has been determined that the spent fuel pool cooling systems are operable to ensure that the 125 F pool temperature will not be exceeded." To illustrate how this will be accomplished, Table I in the November 18, 1983 submittal shows that additional decay time will be imposed before unloading would commence when the service water inlet temperature was 95 F for both maximum normal and maximum abnormal heat loads. Also, the length of the additional decay time will depend on whether one or two cooling trains are operating. The licensee has performed calculations regarding spent fuel pool boiling assuming loss of the pool cooling system. The shortest calculated time to boil under the most adverse conditions is 9.3 hours and an additional 105 hours of boiling would be required before the fuel assemblies will commence to be uncovered. The maximum calculated boiloff rate is 34 gpm which is less than the pool makeup rate of 75 gpm available from the condensate storage and transfer system, and therefore this system is acceptable as the primary makeup source. Further, as a backup makeup water system, 100 gpm is available via the fire protection system from Lake Ontario.

The staff has determined that the 9.3 hours required to reach boiling plus the additional 105 hours of boiling that would be required before the fuel assemblies would commence to become uncovered provides sufficient time to activate either the primary or backup water system in order to prevent the fuel from being uncovered, and is therefore acceptable.

2.4.2 Spent Fuel Cooling

The eight new fixed poison type storage racks located in the south end of the pool will be fabricated from Type 304 stainless steel. They will be freestanding, i.e., unattached or anchored to the pool floor or walls. One rack will have 198 storage locations and the remaining seven will each have 216 storage locations. The gap between storage racks will be 1/4 inch and the clearance between the pool walls and rack will vary from 19.1 inches to 4.0 inches. The licensee stated that no lateral forces will be developed as a result of differences in the pool water temperature with respect to the pool structure and the difference in thermal expansion of the racks with respect to the pool structure. Within the rack, the fuel and fixed poison material are contained in storage boxes. The fuel storage boxes are formed of stainless steel such that two fuel assemblies are housed within one box with a partition. The poison, two 11-1/4 inches wide strips of 0.110 inch thick Boraflex, will be similarly jacketed in Type 304 stainless steel clad boxes that will be placed alongside one side of the fuel containing boxes. The racks will be assembled from combinations of these two types of boxes such that the normal lateral center to center distance between fuel assemblies will be 7.8 inches on one axis and 6.01 inches on the other.

Each rack is supported 11-1/4 inches above the pool floor to form a lower plenum. An analysis performed by the licensee shows that the pool water flow is such that the exit temperature of the pool water will be significantly below the corresponding saturation temperature for the hottest fuel assembly placed in the most adverse location. Then nucleate boiling will not occur. We have reviewed the thermal-hydraulic characteristics of the storage racks and conclude that they are adequate and therefore acceptable.

2.5 Criticality Considerations

Analysis Methods

The spent fuel pool criticality calculations are based on unirradiated fuel assemblies with no burnable poisons which have a maximum fuel enrichment of 3.75 weight percent U-235. This corresponds to a fuel loading of 18.13 grams of U-235 per axial centimeter of fuel assembly. Only the poisoned high density racks in the south half of the pool were analyzed for fuel containing 3.75 weight percent U-235. Previous criticality analysis for the nonpoisoned flux trap racks in the north half of the pool used 15.6 grams of U-235 per axial centimeter and was approved by the staff. This corresponds to 3.0 weight percent U-235 and still remains the limiting average enrichment for fuel placed in the flux trap racks in the north half of the spent fuel pool.

Pickard, Lowe, and Garrick Inc. (PLG) performed the criticality analyses for the spent fuel racks. The PDQ-7 computer code was used for the reactivity determination with four energy group neutron cross sections generated by the LEOPARD code. These codes have been benchmarked against 12 critical experiments performed at Battelle Pacific Northwest Laboratories, seven of which incorporated thin, heavily-absorbing materials. The overall average calculated K_{eff} for these 12 experiments was 0.9931, with a standard deviation value of 0.0011 Δk. Therefore, this benchmarking led to the conclusion that the calculational model is capable of determining the multiplication factor (k_{eff}) of the Nine Mile Point Unit 1 spent fuel racks with a combined LEOPARD/PDQ-7 model bias of +0.0022 Δk uncertainty corresponding to a 95 percent probability at a 95 percent confidence level (95/95).

Spent Fuel Rack Analysis

The criticality of fuel assemblies in the south half of the Nine Mile Point Unit 1 spent fuel pool is prevented by maintaining a minimum separation of 7.805 inches between rows of fuel assemblies and by inserting the neutron absorber, Boraflex, between rows of fuel assemblies. The NRC acceptance criterion for spent fuel storage is that there is a 95 percent probability at a 95 percent confidence level (including uncertainties) that K_{eff} of the fuel assembly array will be less than 0.95 for all storage conditions.

In addition to the calculational method uncertainty mentioned previously, uncertainties and biases due to fuel cell dimensions, pitch between rows of fuel cells, Boraflex loading, fuel pellet density, fuel position, and pool water temperature are included either by using worst case initial conditions or by performing sensitivity studies to obtain the appropriate values. All uncertainties were at least 95/95 probability/confidence values.

Using these methods and assumptions, the nominal k_{eff} of the spent fuel racks in the south half of the spent fuel pool is calculated as 0.9105. The fuel is assumed to be unirradiated with no burnable poison at a higher than expected average enrichment of 3.75 weight percent U-235, corresponding to 18.13 grams of U-235 per axial centimeter. The basic storage rack cell used for the analysis included a fuel bundle wherein the enrichment of each of the 62 contained fuel rods was 3.75 weight percent U-235. In reality, a fuel bundle will have a distribution of fuel rod enrichments rather than a uniform rod enrichment. Therefore, a calculation was also performed for a more realistic fuel assembly with a specific distribution of enrichments which yield an average enrichment of 3.75 weight percent U-235. The K_{∞} of this latter cell was 0.8997 and, therefore, the perturbation to the basic rack cell resulting from a typical realistic enrichment distribution is -0.0108. Since this enrichment is higher than any present design, the particular enrichment distribution selected to represent a typical bundle was based on a reference bundle design with a maximum average planar enrichment of 3.01 weight percent U-235 (fuel bundle P8DRB282 of NEDO-24195). The enrichment of each fuel rod type was increased by the ratio of 3.75/3.01 to obtain the distribution used in the calculation. The pool water temperature was conservatively taken to be 68F as compared to the normal operating temperature of 101F.

Adding the appropriate 95/95 probability/confidence uncertainties and biases yields a value of 0.9307 for the multiplication factor. This meets our acceptance criterion of 0.95.

Accident Analysis

The most limiting accident was found to be the inadvertent placement of a fresh bundle adjacent to a fully loaded rack. The maximum effect of this accident was calculated to be perturbation of +0.0121 k, still resulting in a k_{eff} less than 0.95.

Technical Specifications

Administrative controls will be used to assure that only assemblies with an average enrichment of less than 3.0 weight percent U-235 will be stored in the flux trap racks in the north half of the pool while 3.75 weight percent U-235 assemblies or less will be stored in the poisoned high density racks in the south half of the pool. The Nine Mile Point Unit 1 Technical Specifications have been modified to contain these restrictions on maximum enrichments as a part of this licensing action.

Based on our review, we find that the storage racks meet the requirements of General Design Criterion 62 as regards criticality. Also, we find that any number of fuel assemblies of maximum average enrichment of 3.75 weight percent U-235, which corresponds to 18.13 grams of U-235 per axial centimeter, may be stored in the poisoned high density racks in the south half of the fuel pool. The flux trap racks in the north half of the pool remain limited to assemblies with average enrichments no greater than 3.0 weight percent U-235 (15.6 grams of U-235 per axial centimeter). These findings are based on the following considerations:

1. Calculational methods which have been verified by comparison with experiment have been used.

2. Conservative assumptions have been made about the enrichment of the fuel to be stored and the pool conditions.

3. Credible accidents have been considered.

4. Suitable uncertainties have been considered in arriving at the final value of the multiplication factor.

5. The final effective multiplication factor value meets our acceptance criterion of less than or equal to 0.95.

6. The change to the Nine Mile Point Unit 1 Technial Specifications to contain the two restrictions on maximum enrichment.

We recommend that the administrative controls for the placement of fuel assemblies in the non-poisoned flux trap racks and in the poisoned high density racks be established and incorporated into the plant operating procedures.

2.6 Spent Fuel Pool Water Cleanup Considerations

Description

The spent fuel pool cleanup system is incorporated as a part of the spent fuel pool cooling system. The spent fuel cooling system for the plant consists of two 100% capacity pumps, two heat exchangers, two precoat type filters, two skimmer surge tanks, associated piping, valves and instrumentation. The skimmer surge tanks are designed to remove debris from the pool water and provide pump suction. The precoat filters (mixed bed resin precoat) are designed to remove corrosion products, fission products, and impurities from the pool water. The precoat filters and heat exchangers can be used with either pump for operational flexibility. Both systems can be operated in parallel. Pool water purity is monitored by periodic grab samples for laboratory analysis. Once a week, samples are taken for chemical and radio-chemical analysis. Operational guides for demineralizer resin replacement are: (1) effluent conductivity equals influent conductivity at values above 1 μmho/cm, (2) effluent conductivity exceeds 1 μmho/cm by a significant margin,

(3) differential pressure reaches 25 psi, (4) chlorides exceed 100 ppb and (5) gross gamma activity exceeds 1×10^{-3} µCi/ml.

The licensee indicated that no change or equipment in addition to the spent fuel pool cleanup system is necessary to maintain pool water quality for the increase in fuel storge capacity.

Evaluation

Past experience showed that the greatest increase in radioactivity and impurities in spent fuel pool water occurs during refueling and spent fuel handling. The refueling frequency and the amount of core to be replaced for each fuel cycle, and frequency of operating the spent fuel pool cleanup system are not expected to increase as a result of high density fuel storage. The chemical and radionuclide composition of the spent fuel pool water is not expected to change as a result of the proposed high density fuel storage. Past experience also shows that no significant leakage of fission products from spent fuel stored in pools occurs after the fuel has cooled for several months. To maintain water quality, the licensee has established the frequency of chemical and radiochemical analysis that will be performed to monitor the water quality and the need for spent fuel pool cleanup system demineralizer resin and filter replacement. In addition, the licensee has also set the chemical and radiochemical guidelines to be used in monitoring the spent fuel pool water quality and initiating corrective action. These guidelines are consistent with the reactor coolant Technical Specification water quality requirements.

The facility contains waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation, dated July 1974. There will be no change in the waste treatment system or in the conclusions given in Section 6.1 of the evaluation of these systems because of the proposed modification.

On the basis of the above, we determined that the proposed expansion of the spent fuel pool will not appreciably effect the capability and capacity of the spent fuel pool cleanup system. More frequent replacements of filters and demineralizer resin, if necessary, could offset any potential increase in the pool water as a result of the expansion of stored spent fuel. Thus we have determined that the existing fuel pool cleanup system with the proposed high density fuel storage (1) provides the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool and thus meets the requirements of GDC 61 in Appendix A of 10 CFR Part 50 as it relates to appropriate systems to fuel storage; (2) is capable of reducing occupational exposures to radiation by removing radioactive products from the pool water, and thus meet the requirements of Section 20.1(c) of 10 CFR Part 20, as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the pool water within the filters and demineralizers, and thus meets Regulatory Position C.2.f(2) of

Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the sources; and (4) removes suspended impurities from the pool water by filters, and thus meets Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing crud from fluids through physical action. Therefore, no change to the spent fuel pool cleanup system is required.

2.7 Occupational Radiation Exposure

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for performing the modification is estimated by the licensee to be between 15 and 20 man-rem. If the modification is completed in a single step, the man-rem exposure is expected to decrease slightly as compared to performing this operation in several steps. However, the latter modification method is preferred because the licensee believes that stepwise modification of the pool may result in less man-rem exposure if all the steps are not needed.

The man-rem exposure estimate, as given above, is based on the licensees detailed breakdown of occupational exposure for each phase of the modification considering the man-rem occupational exposure experience of his 1978 SFP modification. He has used this experience as a basis for calculating the exposures expected for each step in his matrix. Consequently, based on his 1978 modification occupational exposure, it is expected that divers operation will account for a significant fraction of the man-rem exposure. However, the licensee is planning on keeping radiation exposures to divers to as low as is reasonably achievable (ALARA) levels by vacuuming the pool floor and other underwater surfaces where such vacuuming will reduce the dose rate, and by keeping the minimum distance between the divers and the nearest spent fuel elements to eight feet. The alternative for performing the diver modification activity is for many people working at the operating deck level using remote handling equipment. This alternative may not achieve a reduction in exposure because of the significantly longer time that may be involved to perform the operation even if in a lower radiation field. Additionally, there would be no guarantee that diver assistance would still not be required because of problems with the remote equipment.

For SFP modification operations that will be performed at the operation deck level, the licensee will keep radiation exposure to personnel working there to ALARA exposure levels by removing radioactive crud deposited on the SFP walls, and by optimizing use of the SFP clean-up filter and demineralizer system to remove insoluable activity in the water. By using the aforementioned techniques, the staff concludes that the SFP modification can be performed in a manner that will ensure ALARA exposures to occupational workers.

The licensee has presented alternative plans for the disposal of the old racks which considered removing and crating intact racks versus removing, cutting and then crating the racks. He is considering two methods of disposal: (1) cutting the old racks into small sections to significantly reduce the volume to be shipped to the burial site or (2) crating the racks whole which will reduce the man-rem exposure involved with disposing of these racks. Cutting the old racks into small sections will permit more efficient packaging in the shipping containers. This will result in a smaller volume of radioactive waste to be disposed of with resulting economic and environmental benefits, e.g., fewer waste shipments and conservation of low level waste burial site space. This will also require that the licensee expend effort to cut the old racks which would result in an increase in occupational exposure. The exposure from the removal, decontamination and packaging of the old racks in the 1978 SFP modification resulted in a 1.2 manrem dose. At this time taking into account alternative disposal costs and exposures, the licensee will make the final decision as to the choice of method of disassembly and disposal of the old racks so that exposures will be kept to ALARA levels.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the spent fuel area, from radionuclide concentrations in the SFP water and deposited on the SFP walls. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in additional exposure will not effect the licensee's ability to maintain individual occupational exposures to ALARA levels and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

3.0 CONCLUSION

We have performed an evaluation of the licensee's proposed modifications based primarily on information provided to us in the licensee's basic supporting document. This document has been revised and supplemented during the course of our review in response to staff questions, and from meetings and discussions with the licensee, and to address new or more refined information regarding the proposed modification.

Our evaluation concludes that the proposed modification of the Nine Mile Point Nuclear Station, Unit 1 spent fuel storage is acceptable because:

(1) The structural and mechanical design for the proposed modification satisfies the applicable requirements of General Design Criteria 2, 4, 61, and 62 of 10 CFR Part 50, Appendix A are acceptable.

(2) The compatibility of the materials and coolant used in the spent fuel storage pool is adequate based on tests, data, and actual service experience in operating reactors. The selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, by having a capability to permit appropriate periodic inspection and testing of components and Criterion 62, by preventing criticality by maintaining structural integrity of components.

(3) The installation of the proposed fuel handling racks can be accomplished safely.

(4) The likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool is sufficiently small that no additional restrictions on load movement are necessary since heavy loads will not be handled over stored spent fuel during reracking and general heavy load handling will be accomplished in accordance with the general guidelines of NUREG-0612.

(5) The cooling system for the spent fuel pool has cooling capacity for normal and abnormal heat loads to maintain pool temperatures within the limits of SRP Section 9.1.3.

(6) The new fixed poison storage racks will adequately permit sufficient natural circulation flow of pool water to suppress nucleate boiling.

(7) The primary and backup sources of makeup water exceed the maximum boil-off rate.

(8) Sufficient time is available to activiate either or both makeup systems before the fuel will commence to become uncovered.

(9) The physical design of the new storage racks will preclude criticality for any credible moderating condition.

(10) The existing SFP cleanup system is adequate for the proposed modification.

(11) The conclusions of the evaluation of the waste treatment systems are unchanged by the modification of the spent fuel pool.

(12) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the spent fuel pool would be negligible.

We conclude, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public.

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO. 54 TO DPR-63

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT NUCLEAR STATION, UNIT NO. 1
DOCKET NO. 50-220

1.0 Introduction and Discussion

The spent fuel storage capacity at the Nine Mile Point Nuclear Station, Unit 1 was originally 800 BWR fuel assemblies, or storage for approximately 1.5 cores from the unit. This capability was later increased to a maximum of 1140 BWR fuel assemblies. This limited storage capability was in keeping with the expectation generally held in the industry that spent fuel would be kept onsite for a period of 3 to 5 years and then shipped offsite for reprocessing and recycling of the fuel.

Reprocessing of spent fuel did not develop as had been anticipated, however, and in September 1975, the Nuclear Regulatory Commission (NRC, the Commission) directed the NRC staff (the staff) to prepare a Generic Environmental Impact Statement (GEIS, the Statement) on spent fuel storage. The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. The Statement would consider alternative methods of spent fuel storage as well as the possible restriction or termination of the genration of spent fuel through nuclear power plant shutdown.

A Final Generic Environmental Impact statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575), Volumes 1-3 (the FGEIS) was issued by the NRC in August 1979. In the FGEIS, consistent with the long range policy, the storage of spent fuel is considered to be interim storage, to be used until the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion of onsite fuel storage capacity by modification of the existing spent fuel pools. Applications for fifty such spent fuel capacity increases have been reviewed and approved. The finding in each case has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage pool designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommends that licensing reviews be done on a case-by-case basis to resolve plant specific concerns.

In addition to the alternative of increasing the storage capacity of the existing spent fuel pool, other spent fuel storage alternatives are discussed in detail in the FGEIS. The finding of the FGEIS is that the environmental impact costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of the various alternatives reflect the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, that of shutting down the reactor when the spent fuel storage capacity is field, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Impact Appraisal (EIA) addresses the environmental concerns related only to expansion of the Nine Mile Point Unit 1 spent fuel storage pool. Additional discussion of the alternatives to increasing the storage capacity of existing spent fuel pool is contained in the FGEIS.

1.1 Description of the Proposed Action

In their submittals of March 22 and December 21, 1978, the Niagara Mohawk Power Corporation (the licensee) proposed to increase the licensed total storage capacity of the spent fuel pool (SFP) at Nine Mile Point Nuclear Station Unit 1 (NMP-1) from 1984 with several options to a maximum of 3009 fuel assemblies.

In their submittal dated June 24, 1983 the licensee selected an option in which the maximum licensed capacity would be storage capacity for 2776 BWR fuel assemblies consisting of 1066 flux trap spaces and 1710 poisoned spaces. The 1066 flux trap racks would remain in the north half of the pool and the existing racks in the south half of the pool would be replaced with up to 1710 poisoned spaces in high density racks. This would provide storage for spent fuel generated at Nine Mile Point - 1 while maintaining full core off load capability through the 1994 refueling outage.

The environmental impacts of the Nine Mile Point - 1 facility, as designed, were considered in the NRC's Final Environmental Statement (FES) issued January 1974 relative to the continuation of construction and operation of the facility. The licensee was later authorized to increase the storage capacity from 800 to 1140 by our Safety Evaluation dated March 5, 1976. This is the third proposed SFP modification for NMP-1. The second, which was evaluated in the Safety Evaluation and Environmental Impact Appraisal supporting Amendment 21 to the license dated January 27, 1978. That action increased the licensed storage capacity of the SFP from 1140 to 1984 fuel assemblies.

In this EIA we have evaluated any additional environmental impacts which are attributable to the proposed increase proposed by the licensee in their March 22, and December 21, 1978 submittals in the SFP storage capacity for the Station.

1.2 Need For Increased Storage Capacity

A spent fuel storage pool is currently provided at Nine Mile Point - 1 with 1066 spaces in high density flux trap racks in the north half of the spent fuel pool and 520 spaces in the existing original racks in the south half of the pool. With the exception of 22 spaces, all spaces in the north half of the pool are full. Twenty four fuel assemblies from the north half of the pool will be re-inserted into the reactor core. Therefore, a total of 46 spaces in the north half and 520 spaces in the south half will be available for fuel storage. During the 1984 refueling a total of 200 fuel assemblies will be discharged into the pool. If the proposed modification is not completed, the ability to fully discharge the reactor core would be lost following the upcoming refueling outage. The proposed modification would be full core discharge capability through the 1994 refueling outage.

1.3 Fuel Reprocessing History

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shutdown in 1972 for alterations and expansion; in September 1977, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The pool is on land owned by the State of New York. NFS's lease with the State of New York expired in 1980 and their license has been suspended. The State of New York has requested the utilities who own the spent fuel presently stored in the pool to remove it. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate. The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois is licensed to store spent fuel. On May 4, 1982, the license held by GE for spent fuel storage activities at its Morris Operation was renewed for another 20 years. GE is not accepting any additional spent fuel for storage at this facility.

2.0 THE FACILITY

The principal features of the spent fuel storage and handling at Nine Mile Point - 1 as they relate to this action are described here as an aid in following the evaluations in subsequent sections of this environmental impact appraisal.

2.1 The Spent Fuel Pool (SFP)

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core; also, they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

2.2 SFP Cooling System

The spent fuel and cooling system (SFPCS) at the Nine Mile Point Nuclear Station, Unit 1 consists of two pumps in parallel, with a pump and heat exchanger in series. The heat removal design capability of each heat exchanger is 6.8×10^6 Btu/hr at 116F and 8.3×10^6 Btu/hr at 125F.

Heat is transferred from the spent fuel pool cooling system to the reactor building closed cooling water system. The reactor building closed cooling water system, in turn, transfers heat to the service water system. The RHR system is also a closed system cooled by service water. The service water system is a once-through cooling system in which strained water from Lake Ontario is supplied from pumps in the intake structure and returned to the lake after removing heat from a number of systems, including the reactor building closed cooling water and the RHR systems.

2.3 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the NRC"s Final Environmental Statement (FES) dated January 1974. There will be no change in the waste treatment systems described in Section 3.5 of the FES because of the proposed modification.

2.4 Spent Fuel Pool Cleanup System

The SFP cooling and cleanup system consists of two surge tanks, two circulating pumps, two heat exchangers, two precoat filter-demineralizers and the required piping, valves and instrumentation. The pumps draw water from the surge tanks and discharge it through the heat exchangers and the filter-demineralizers to the SFP. One loop with a single filter-demineralizer and heat exchanger is used normally. The second loop is on standby available to operate in parallel with the other loop to provide additional cooling and filtration.

3.0 ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

3.1 Nonradiological Consequences of the Proposed Action

The nonradiological environmental impacts of Nine Mile Point - 1, as designed, were considered in the FES issued January 1974. Increasing the number of assemblies stored in the existing fuel pool will not cause any new nonradiological environmental impacts not previously considered. The amounts of waste heat emitted by the unit as a result of the proposed increased spent fuel storage capacity will increase slightly (less than one percent), but will result in no measurable increase in impacts upon the environment.

3.2 Radiological Consequences of the Proposed Action

3.2.1 Introduction

The potential offsite radiological environmental impact associated with the expansion of spent fuel storage capacity at Nine Mile Point - 1 has been evaluated.

During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54, which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominantly nonvolatile at the temperature conditions that exist in pool storage. The primary impact of such nonvolatile radioactive nuclides is their contribution of radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of the radionuclides in the pool water apepar to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent pool during refueling operations), or crud dislodged from the surface of the spent fuel during transfer from reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably.

A few weeks after refueling, the spent fuel cools in the pool so that the fuel cladding temperature is relatively cool, approximately 180F. This substantial temperature reduction reduces the rate of release of fission products from the fuel pellets, and decreases the gas pressure in the gap between pellets and cladding, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on operational reports submitted by licensees, and discussions with storage facility operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite spent fuel pool, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

3.2.2 Radioactive Material Released to the Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is not significant release of fission products from defected fuel. However, we have conservatively estimated that for this proposed SFP modificaiton an additional 23 curies per year of Krypton-85 may be released from the SFP when the modified pool is filled from 1984 to 3009 spent fuel assemblies. This increase would result in an additional total body dose of less than 0.0001 mrem/year to an individual at the site boundary. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less then 0.0003 man-rem/year. This is small compared to the fluctuations in the annual dose this population would receive from natural background radiation. This exposure represents an increase of less than 0.2% of the exposure from the plant evaluated in the FES. Thus, we conclude that the proposed modification will not have any significant impact on exposures offsite.

We have also conservatively estimated the additional curies per year of Krypton-85 that may be released from the SFP when the modified pool is completely filled from 1140 to 3009 fuel assemblies. The 140 fuel assemblies is the original licensed capacity of the NMP-1 SFP. The licensee's first proposed SFP modification which increased the licensed storage capacity of the SFP from 1140 to 1984 fuel assemblies was evaluated in the Environmental Impact Appraisal dated January 27, 1978, for NMP-1. This estimate, 56 curies per year Krypton-85, is the maximum additional annual amount of gaseous activity that may be released from the NMP-1 SFP because the capacity of the SFP has been increased above the original licensed storage capacity of 1140 assemblies. This increase would result in an additional annual total body dose to an individual at the site boundary and to the population around the plant out to 50 miles is also less than 0.0001 man-mrem/year and 0.0003 man-rem/year, respectively, above these exposures given in the NMP-1 FES. These exposures are also small compared to the fluctuations in the annual dose this population receives from background radiation and are also less than 0.2% of the exposures from the plant evaluated in the NMP-1 FES. Thus, we conclude that the proposed modification of the SFP will not have any significant impact on offsite exposures.

Assuming that the spent fuel will be stored on site for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Storage of additional spent fuel assemblies in the pool is not expected to increase the bulk water temperature during normal refuelings above the 125F used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modification from that previously evaluated in the FES.

Most airborne releases from the plant result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than the spent fuel pool. Therefore, even if there were a slightly higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increase in stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If levels of radioiodine become too high, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. The plant radiological effluent Technical Specifications, which are not being changed by this action, restrict the total releases of gaseous radioactivity from the plant including the SFP.

3.2.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the filter-demineralizer and by decay of short-lived isotopes. The activity is high during refueling operations while reactor coolant water is introduced into the pool and decreases as the pool water is processed through the filter-demineralizer. The increase of radioactivity, if any, should be minor because the additional spent fuel to be stored is relatively cool, thermally, and radionuclides in the fuel will have decayed significantly.

While we believe that there should not be an increase in solid radwaste due to the modification, as a conservative estimate, we have assumed that the amount of solid radwaste may be increased by 66 cubic feet a year from the filter-dimeralizer over that for the SFP with the originally licensed capacity of 1140 fuel assemblies. The annual amount of solid waste shipped from the site was about 18,300 cubic feet for 1972 to 1977. If the storage of additional spent fuel does increase the amount of solid waste from the SFP purification systems by about 66 cubic feet per year, the increase in total waste volume shipped would be less than 0.4% and would not have any significant environmental impact.

The present spent fuel racks to be removed from the SFP because of the proposed modification are contaminated and will be disposed of as low level solid waste. The licensee has estimated that less than 14,300 cubic feet of solid radwaste will be removed from the plant because of the proposed modification. This includes the solid radwaste shipped from the plant because of the 1978 modification of the SFP. Therefore, the total waste shipped from the plant should be increased by less than 2% per year when averaged over the lifetime of the plant. This will not have a significant environmental impact.

3.2.4 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. The amount of radioactivity on the SFP filter-demineralizer might slightly increase due to the additional spent fuel in the pool, but this increase of radioactivity should not be released in liquid effluents from the plant. The plant radiological effluent technical specifications, which are not being changed by this action, restrict the total release of radioactivity in liquid effluents from the plant.

The filter-demineralizer resins are periodically flushed with water to the solid waste system and are not regenerated. The water used to transfer the spent resin is decanted from the tank and returned to the liquid radwaste system for processing. The soluble radioactivity will be retained on the resins. If any activity should be transferred from the spent resin to this flush water, it would be removed by the liquid radwaste system.

Leakage from the SFP would be collected in the reactor building floor drain sumps. The leakage would then be transferred to the liquid radwaste system and processed by the system before any water is discharged from the plant. There have not been signs of leakage from the pool.

3.2.5 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for the entire operation is estimated by the licensee to be between 15 and 20 man-rem. We consider this to be a reasonable estimate because it is based on relevent experience of dose rate measurements and occupancy factors for individuals performing the same specific jobs during the 1978 modification of the NMP-1 SFP. This operation is expected to be a small fraction of the total man-rem burden from occupational exposure per year.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for occupancy times and dose rates in the spent fuel pool area. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

3.2.6 Impacts of Other Pool Modification

As discussed above, the additional environmental impacts in the vicinity of NMP-1 resulting from the proposed modification are very small fractions (less than 1%) of the impacts evaluated in the NMP-1 FES. These additional impacts are too small to be considered anything but local in chapter.

James A. FitzPatrick Nuclear Power Plant (FitzPatrick) is located on the same site as NMP-1. By letter dated July 26, 1978, the Power Authority of the State of New York proposed increasing the spent fuel storage capacity at FitzPatrick. Operation of FitzPatrick was evaluated in the FitzPatrick Final Environmental Statement dated March 1973.

The impact of any environmental significance at NMP-1 from the proposed SFP modification at FitzPatrick is the additional gaseous effluent from the FitzPatrick SFP modification. We have conservatively estimated an additional 99 curies per year of Krypton 85 may be released from FitzPatrick when its modified pool is completely filled. This additional Krypton 85 would result in an additional total body dose, that might be received by an individual near NMP-1 or by the estimated population within a 50 mile radius, of less than 0.001 mrem/year and 0.005 man-rem/year, respectively.

Summing the additional exposures resulting from the SFP modifications at both NMP-1 and FitzPatrick shows the additional total body dose that might be received by an individual and by the estimated population out to 50 miles is less than .0011 mrem/year and .0053 man-rem/year, respectively. These summed exposures are small compared to the fluctuations in the annual dose this population receives from natural background radiation and represent an increase of less than 0.1% of the combined exposures evaluated in the FitzPatrick FES and the NMP-1 FES. These estimates are not significant.

Based on the above, we conclude that a SFP modification at any other facility should not significantly contribute to the environmental impact of NMP-1 and that the SFP modification should not contribute significantly to the environmental impact of any other facility.

3.3 Environmental Impact of Spent Fuel Handling Accidents

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the FES dated January 1974.

The heavy load handling operations associated with the installation of the new poison type racks in the south end of the pool will be accomplished without handling of heavy loads over stored spent fuel. Further, general heavy load handling operations will be accomplished in accordance with the general guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Therefore, we have concluded then that the likelihood of a heavy load handling accident is sufficiently small that the proposed modifications are acceptable, and no additional restriction on load handling operations in the vicinity of the SFP are required.

3.4 Radiological Impacts to the Population

The proposed increase of the storge capacity of the SFP will not create any significant additional radiological effects to the population. The additional total body dose that might be received by an individual at the site boundary, and by the estimated population within a 50-mile radius, is less than 0.0001 mrem/yr and 0.0003 man-rem/yr, respectively. These doses are small compared to the fluctuations in the annual dose this population receives from background radiation. This population dose represents an increase of less than 0.2 percent of the dose previously evaluated in the FES for the Nine Mile Point Nuclear Station, Unit 1. We find this to be an insignificant increase in dose to the population resulting from the proposed action.

4.0 Summary

The findings contained in the Final Generic Environmental Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, (the FGEIS) issued by the NRC in August 1979, were that the environmental impact of interim storage of spent fuel was negligible, and the cost of the various alternatives reflect the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in spent fuel pool designs, the FGEIS recommended licensing spent fuel pool expansions on a case-by-case basis. Expansion of the spent fuel storge capacity at Nine Mile Point Nuclear Station, Unit 1 does not significantly change the radiological impact evaluated by the NRC in the FES issued in January 1974. As discussed in Section 3.4 of this EIA, the additional total body dose that might be received by an individual at the site boundary or the estimated population within a 50-mile radius is less than 0.0001 mrem/yr and 0.003 man-rem/yr respectively, and is less than the natural fluctuations in the dose this population would receive from background radiation. The occupational exposure for the modifications of the SFP is estimate by the licensee to be 15 to 20 man-rem. This is conservative. Operation of the plant with additional spent fuel in the SFP is not expected to increase the occupational radiation exposure by more than one percent of the total annual occupational exposure at the two units.

5.0 Basis and Conclusion For Not Preparing an Environmental Impact Statement

We have reviewed the proposed modifications relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6. We have determined, based on this assessment, that the proposed license amendment will not significantly affect the quality of the human environment. Therefore, the Commission has determined that an environmental impact statement need not be prepared and that, pursuant to 10 CFR 51.5(c), the issuance of a negative neclaration to this effect is appropriate.

Principal Contributors:

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Dated: February 1, 1984

U. S. NUCLEAR REGULATORY COMMISSION

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-220

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

AND NEGATIVE DECLARATION

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 54 to Facility Operating License No. DPR-63, issued to Niagara Mohawk Power Corporation (the licensee), which revised the Technical Specifications for Operation of the Nine Mile Point Nuclear Station, Unit No. 1 (the facility) located in Oswego County, New York. The amendment is effective as of the date of issuance.

The amendment authorizes changes to the Technical Specifications to allow an increase in the spent fuel storage capacity to a maximum of 2776 assemblies.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment in connection with this action was published in the FEDERAL REGISTER on November 7, 1978 (43 FR 51883). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Impact Appraisal related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated January 1974.

For further details with respect to the action see (1) the application for amendment dated March 22, 1978, December 20 and 21, 1978, February 26, 1981, June 24, August 5, October 5, October 26, November 18 and December 21, 1983, and January 3, 1984 (2) Amendment No. 54 to License No. DPR-63, (3) The Commission's Safety Evaluation, and (4) The Commission's Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the State University College at Oswego, Penfield Library - Documents, Oswego, New York 13126. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 1st day of February, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing