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SUBJECT: Suppl to 780321 application to amend License DPR-63,revising Tech Spec Sections 5.5 & 5.6 re spent fuel pool.Also clarifies info in 781220 & 21 ltrs & forwards spent fuel pool structural analysis.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

June 24, 1983

Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Gentlemen:

Our proposed license amendment dated March 21, 1978 provided information on the spent fuel pool capacity expansion at Nine Mile Point Unit 1. Additional information was also provided in our letters of December 20 and 21, 1978. The attachments to this letter are provided to clarify and supplement that submittal. Also provided are responses to questions raised by members of your staff as well as revised Technical Specification Sections 5.5 and 5.6 and the Spent fuel Pool Structural Analysis.

In order to maintain core off-load capability after the 1984 refueling and maintenance outage, approval of this capacity expansion is required by September 30, 1983.

Very truly yours,

C. V. Mangan
Vice President
Nuclear Engineering and Licensing

CVM/MGM:bd

Attachs.

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NIAGARA MOHAWK POWER CORPORATION
DOCKET NO. 50-220
DPR-63

SUPPLEMENTAL SUBMITTAL

Nine Mile Point Unit 1

Spent Fuel Pool Modification

June 1983

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1.0 SUMMARY

The information presented herein supplements the March 21, 1978 submittal. That submittal outlined seven potential storage matrices. Based on the future need for storage capacity, economics and ALARA, storage matrix 6, as summarized on Table 1 of the March 21, 1978 submittal, will be utilized at Nine Mile Point Unit 1. This storage option consists of 2776 spent fuel storage locations containing both non-poison and poison density racks. The north half of the spent fuel pool currently contains 1066 storage spaces with eight non-poison flux trap density racks. These racks, installed in 1978, will remain in the spent fuel pool. The south half of the spent fuel pool currently contains 520 storage spaces with 26 original design spent fuel storage racks. This submittal proposes to remove those racks and replace them with 1710 storage spaces with eight poison high density racks.

The analyses of the Spent Fuel Pool Modification (Phase II) encompassed the integrity of the entire pool and related systems including effects of the Phase I racks, with a full pool having 2,776 fuel assemblies in storage. This Spent Fuel Pool Modification, related design objectives and analyses generally comply with the guidance provided in the "OT Position for Review and Acceptance of Spent Fuel Handling and Applications."

1.1 History-Phase-I

Phase I consisted of the installation of 1,066 spent fuel storage locations (eight racks) of the non-poison flux trap design, described in our March 21, 1978 submittal. Each rack is bolted to a pre-placed rack base in the pool. The racks cantilever from the rack bases with no lateral supports at the top. The rack bases are provided with leveling screws. Gaps at lateral seismic supports allow for thermal growth while still providing lateral restraint. Lateral loads in the east, west and north directions are transmitted through the bases to lateral restraints located on the east, west and north pool walls. Lateral loads in the south direction are transmitted from the rack bases to restraints located on the floor and supported by the existing pool floor swing bolt brackets. The racks are allowed to tip and impact on the pool floor. Installation of Phase I was completed in July 1978.

The racks in the north half of the pool are isolated from the racks in the south half of the pool. There is no mechanical connection or bearing between the north and south half fuel racks. Hence, lateral load cannot be transferred between the north and south half.

1.2 Proposed Modification - Phase II

Phase II consists of the installation of 1,710 spent fuel storage positions (eight racks) of the poison type which use Boraflex as the neutron absorber.

1.2 Proposed Modification - Phase II (Continued)

Each rack is supported by four floor bearing pedestals. The racks are not interconnected at the base or top and are free to slide and tip within the confines of the lateral restraints. A 1/4 inch gap will be established at installation between the racks and the wall bearing pads in the east, west and south directions. Lateral loads in the east, west and south directions are transmitted through individual rack bottom plates to bearing pads on the east, west and south pool walls. Lateral loads in the north direction are transmitted to existing swing bolt brackets through a support beam located under each rack. Design and installation requirements provide for direct contact with seismic restraints in this direction. Two additional lugs will be welded to the pool liner for lateral support in the north direction for the racks located on the east edge of the pool. These lugs are required since swing bolts do not exist at these locations. For final pool configuration and seismic restraint and pedestal details, see Figures 1, 2, 3, 4 and 5.

The north half and south half fuel racks are similar in construction. Boxes are full seam welded and fusion spot welded together along their long axes (see Figure 6). The flux trap design contains water boxes, whereas the poison design contains boxes housing Boraflex poison inserts (see Figure 7). All racks are constructed of 304 stainless steel.

Our current plans call for the installation of six racks (1,296 positions) and two Work Platforms (see Figure 8). The Work Platforms serve to provide space for general spent fuel pool work, space and support for control rod racks and westerly lateral restraints for the six rack modules. The Work Platforms will be used until full off-core loading capability calls for additional storage positions. At that time, the remaining two spent fuel storage racks will be installed.

The six racks will be temporarily unrestrained in a westerly direction when a change out of the work platforms occurs. Under an Operating Basis Earthquake condition, the racks will not slide to the extent of closing the 1/4 inch gaps between racks. Impacting will not occur. Therefore, safety and structural stability during a seismic event for the operating conditions will not be compromised. Because of the low probability of a Safe Shutdown Earthquake event happening during the short change out period, the Safe Shutdown Earthquake condition was not considered as a credible design consideration.

1.3 Nuclear Regulatory Commission Documents

The following Nuclear Regulatory Commission documents provided guidance for the design and analyses of the components of the Spent Fuel Pool Modification with exceptions noted:

Regulatory Guide 1.29 - "Seismic Design Classification"

Regulatory Guide 1.60 - "Design Response Spectra for Seismic Design of Nuclear Power Plants"

Regulatory Guide 1.61 - "Damping Valves for Seismic Design at Nuclear Power Plants"

Regulatory Guide 1.92 - "Combining Modal Responses and Spatial Components in Seismic Response Analysis"

Regulatory Guide 1.122 - "Development of Floor Design Response Spectra for Seismic Design of Floor - Supported Equipment or Components"

Regulatory Guide 1.124 - "Service Limits and Loading Combinations for Class 1 Linear - Type Component Supports"

Regulatory Guide 1.142 - "Safety - Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)"

Exception: The recommendations of paragraph 9 regarding more conservative load factors was not followed in the analysis of the Spent Fuel Pool Structure Analysis, see discussion in Section 5.2.

NUREG-0800 - "Standard Review Plan"

Section 3.8.4 - "Other Seismic Category I Structures"

Exception: See exception to Regulatory Guide 1.142.

Appendix D to Section 3.84 - "Technical Positions on Spent Fuel Pool Racks"

Exception: See exception to Regulatory Guide 1.142.

Section 9.1.2 - "Spent Fuel Storage"

Section 9.2.5 - BTP ASB 9-2 "Residual Decay Energy for Light - Water Reactors for Long Term Storage"

1.4 Conclusion

On the basis of the information presented in the March 21, 1978 submittal and this Supplement, Niagara Mohawk concludes that the proposed modification will provide safe storage for up to 2,776 fuel assemblies. As indicated in this submittal:

a) The proposed modification accounts for rack mechanical effects on the pool and its environment with adequate margins of safety.

b) The proposed modification accounts for reactivity effects on the racks, spent fuel pool systems and the overall environment with adequate margins of safety.

c) The proposed modification accounts for seismic effects on the racks and the spent fuel pool with adequate margins of safety.

d) The proposed modification accounts for loading effects on the spent fuel pool and liner with adequate margins of safety.

e) The proposed modification accounts for thermal-hydraulic effects on the racks, spent fuel pool and its systems, with adequate margins of safety.

f) The proposed modification reduces the overall margins of safety for the spent fuel pool due to increased fuel storage capacity. However, analyses show that resulting stresses are within applicable codes and other requirements.

g) The proposed modification meets the intent of the Nuclear Regulatory Commission's recommendations pertaining to Spent Fuel Pool Modifications.

2.0 RACK MECHANICAL ANALYSIS

The high density poison racks have been designed to maintain stored spent fuel in a structural and physical array that protects spent fuel assemblies and prevents criticality under credible storage conditions. The analysis and design encompasses credible storage conditions.

2.1 Design Criteria

The spent fuel storage racks have been designed to comply with NRC Position Papers, Regulatory Guides 1.29 and 1.124, Appendix "D" of the Standard Review Plan Section 3.8.4 and ASME "Boiler and Pressure Vessel Code" Section III Subsection NF, 1977 Edition, Summer 1979 Addendum.

2.2 Methods of Analyses

The analyses of the spent fuel racks used loads developed from the Seismic Analysis and were applied to critical portions of the rack and supporting structure. Stresses (axial, shear, bending and torsion) were then evaluated and compared to allowables to determine structural adequacy.

2.3 Loads and Loading Combinations

Appendix D of Standard Review Plan Section 3.8.4 defines the load combinations and acceptance limits for each combination. These are listed in Table 1. The abbreviations and their significance in this analysis are as follows:

a. D - Dead loads consisting of the weight of the racks, supports and the entrained pool water.

b. L - Live loads consisting of fuel assemblies.

c. T_o - Temperature effects and loads during normal operating or shutdown conditions. 125°F was used for evaluating material properties for the load combinations containing this term.

d. T_a - Temperature effects at the highest temperature associated with the postulated abnormal design conditions. For conservatism, the maximum temperature of the racks was assumed to be 200°F.

e. E - Operating basis earthquake (OBE).

f. E' - Safe shutdown earthquake (SSE).

2.3 Loads and Loading Combinations (Continued)

g. P_f - Loads caused by a stuck fuel assembly in any rack location.

Maximum upward load	=	2,000 lbs.
Maximum downward load	=	1,000 lbs.

h. F_d - Loads due to postulated fuel handling accident. These include:

1. Straight drop and an inclined drop of a fuel assembly on the top of the rack from a 33 inch height.
2. A straight drop of a fuel assembly through an individual cell to the bottom of a rack.

The racks and pedestals are considered linear-type supports and were designed and analyzed in accordance with ASME "Boiler and Pressure Vessel Code," Section III; Appendix I, Appendix XVII, Appendix F and Subsection NF. In Subsection NF, paragraph NF-3132.3 states that linear-type supports shall be analyzed "based on the maximum stress theory in accordance with the rules of paragraph NF-3230 and Appendix XVII."

The following limits from Subsection NF, paragraph NF-3231.1, apply for Level A, B, and D service loading conditions:

a. For Level A and B Loadings (Normal and Upset), normal limits of Appendix XVII - 2000 apply. For stresses resulting from constraints of free end displacements, three times the normal limits are allowed.

b. For Level D Loadings (Faulted), Appendix F, Paragraph F-1370 allows the normal limits of Appendix XVII - 2000 to be increased by a factor of 1.2 S_y/F_t not to exceed 0.7 S_u/F_t. F_t is the allowable tensile stress from XVII-2211 which equals 0.6 S_y. Since S_u = 71 KSI and S_y = 25 KSI at 200°F, then 1.2 S_y/F_t = 2.0 and 0.7 S_u/F_t = 3.3. Paragraph 5(b) of Standard Review Plan Section 3.8.4 limits this factor to 1.6. Appendix D to section 3.8.4, which deals specifically with fuel racks, does not limit this factor. To ensure conservatism, the lowest factor (1.6) was adopted.

As recommended by Section 3.8.4 of the Standard Review Plan, the limits of NF-3231.1 were amended by the requirements of Regulatory Guide 1.124.

2.4 Material Properties

The storage racks and their supports are fabricated from stainless steel, ASTM-A240, Type 304 plate. Properties are evaluated at both T_o (125°F) and T_a (200°F). Values for material properties were taken from Tables I-2.2 and I-3.2 of ASME, Section III, Division 1, Appendix I.

2.5 Conclusion

The analysis and design meets the design criteria outlined in Section 2.1, including a fuel drop accident which would not adversely affect the sub-critical array of the fuel assemblies.

The analysis also included the loading effects from the racks and seismic restraints on the pool liner and liner supports. Shear loads through attached swing bolts and seismic shear lugs and bearing loads from wall lateral seismic restraints and rack support pedestals inducing stresses in the pool liner were evaluated in accordance with Appendix XVII of Section III of the ASME Code. The attachments, restraints, pedestals and liner plate were established to be within the code limits.

3.0 CRITICALITY ANALYSIS

Nuclear criticality analyses have been performed to demonstrate that the Phase II poison design spent fuel rack system is acceptably subcritical (that is the effective multiplication factor, K_{eff}, is less than 0.95). These analyses included consideration of credible normal and abnormal operating occurrences.

The criticality analyses presented in the March 21, 1978 submittal were based upon a maximum fuel enrichment of 3.00 weight percent U-235. Because considerable margin existed to the 0.95 K_{eff} allowable and based upon recognition of the potential need for future enrichments exceeding this value, the Phase II poison design spent fuel rack system has been analyzed for a maximum enrichment of 3.75 weight percent U-235. This is higher than any anticipated reload enrichment for Nine Mile Point 1.

The boron - 10 loading in the Boraflex sheets is unchanged at 0.0217 grams per square centimeter of cross sectional area from the March 21, 1978 submittal. This loading was applied to the nominal Boraflex thickness of 0.095 $\pm$ .015 inches. It was found that increasing the thickness up to a maximum value of 0.110 inches while maintaining a minimum loading of 0.0217 gr B^{10}/cm^2 resulted in the most limiting K-infinity so these assumptions were incorporated in the calculations.

3.1 Analytical Technique

The Leopard computer program was used to generate four group macroscopic cross sections for input to the PDQ-7 computer program. Leopard calculates the neutron energy spectrum over the entire energy range from thermal up to 10 MeV and determines averaged cross sections over appropriate energy groups. The PDQ-7 program is used in the final predictions of the reactivity of the spent fuel storage rack design. The PDQ-7 calculations were performed in four energy groups and took into account the significant geometric details of the fuel assemblies, fuel boxes and major structural components.

The geometry used for most of the calculations was a "basic cell" which represented one-quarter of the area of a repeating array of a single box consisting of two fuel assemblies and one poison cell (See Figure (9). The basic cell results were then used to calculate the reactivity of an infinite array of uniform spent fuel storage racks.

3.2 Leopard/PDQ-7 Calculational Accuracy

As a specific demonstration of the accuracy of the calculational model, the combined Leopard/PDQ-7 model was used to calculate fourteen measured just critical assemblies. These calculations permitted the determination of the calculational uncertainty which must be accounted for in the spent fuel rack criticality analysis. These combined Leopard/PDQ-7 calculations resulted in a calculated average K_{eff} of 0.9928 with a standard deviation of 0.0012ΔK. Based upon these results, the Leopard/PDQ-7 model was shown to have an accuracy of better than 0.010ΔK at the 95 percent confidence level.

3.3 Analyses, Non-Accident

Criticality analyses were conducted to determine the K_{eff} of the spent fuel storage rack design. The nominal or best estimate case was first analyzed. Deviations from the nominal case were then considered (e.g. consideration of manufacturing tolerances, positional uncertainties, calculational uncertainty and water temperature variation). These perturbations were made to find the worst case, non-accident K_{eff}.

3.3.1 Nominal Case

The analyses for the nominal case considered fresh fuel enriched to 3.75 weight percent U-235 at 95 percent theoretical density in a pool of 68°F water. The basic cell pitch is 7.805 inches between rows. This produced a 0.9105 infinite multiplication factor. A detailed one dimensionsal model was then used to calculate the reactivity effect of the spacing between fuel rack assemblies. The adequacy of the calculational mesh was verified by comparison to an identical calculation with finer calculational mesh, (two times the number of mesh intervals in each direction). The finer mesh calculation resulted in a 0.0003ΔK decrease. Including the Leopard/PDQ-7 model bias and the minimum required bundle burnable poison loading for fresh fuel, the best estimate K_{eff} was found to be 0.8962.

3.3.2 Worst Case

There were a number of tolerances and uncertainties which result in perturbations to K_{eff} that were considered in the criticality analysis if their impact were positive (i.e. resulting in increasing K_{eff}). The reactivity effects of all such positive perturbations were statistically combined to find a single worst case positive perturbation. The worst case positive perturbation value was combined with the nominal case K_{eff} value (which included the Leopard/PDQ-7 modeling bias) to determine the maximum possible multiplication factor.

Among the uncertainties considered were: fuel cell dimensions, pitch between rows of fuel cells, minimization of neutron absorption in the Boraflex poison, minimization of neutron absorption in the rack's stainless steel structural members, variation in the fuel pellets theoretical density, variation of the fuels' position within fuel cells, the 2 sigma uncertainty (corresponding to a 95/95 confidence level) found from the calculational model benchmarking and pool temperature variations.

3.3 Analyses, Non-Accident (Continued)

The conservatively calculated reactivity of the spent fuel rack fully loaded with unirradiated bundles of 3.75 weight percent U-235 (18.13 grams/axial centimeter of assembly) and no burnable poisons, at the most limiting pool temperature of 68°F, including conservative allowances for manufacturing and calculational uncertainties, was found to be 0.9307. When minimum credit is taken for the burnable poison loading, K_{eff} is found not be exceed 0.9007 for fresh, unburned fuel.

3.4 Accident Analysis

The design of the spent fuel rack physically precludes the possibility of a dropped fuel bundle occupying a position below the top of the rack except for a normal location. The reactivity effect of a dropped bundle resting on top of the racks was analyzed and found to increase K_{eff} by 0.0024ΔK. This results from a slight reduction in axial leakage due to the dropped bundle.

The effects of inadvertently placing a fresh bundle adjacent to a fully loaded rack were found to be most limiting. The maximum effect of this accident was found to be a perturbation of +0.0121ΔK. For this case, even assuming the effect is added to the worst case non-accident K_{eff} value without any burnable poison, the resultant K_{eff} is less than 0.95.

3.5 Conclusion

The Leopard/PDQ-7 modeling technique is a well founded method of calculating criticality. The analyses performed have shown that under all credible normal and abnormal operating conditions, the spent fuel poison rack system is more than adequately subcritical (i.e. the maximum value of K_{eff} is considerably less than 0.95).

The analyses conservatively take credit for minimum neutron absorption in the poison material and stainless steel used in the racks.

4.0 SEISMIC ANALYSIS

For the review of the structural adequacy of the subject racks deadweight, Operating Basis Earthquake and Safe Shutdown Earthquake loads were generated. These resulting loads were then used to prepare a detailed stress analyses. Since stored fuel will impact their supporting racks during a seismic event and the racks were assumed to impact the lateral seismic restraints, the non-linear behavior of the racks and fuel assemblies were considered.

The loadings considered in this evaluation were as follows:

a) Submerged weights of fuel storage racks;

b) Submerged weights of fuel bundles with and without channels;

c) Seismic loadings (both Operating Basis Earthquake and Safe Shutdown Earthquake) due to the acceleration time history at the spent fuel pool floor.

4.1 Material Properties

The new spent fuel racks and ancillary parts are fabricated from Type 304 stainless steel.

The 304 stainless steel rack material properties used in the seismic analysis were:

Density	=	501 PCF
Young's Modulus	=	28×10^6 PSI
Shear Modulus	=	10.7×10^6 PSI

In addition, fuel bundles containing cladding and channels constructed of Zircaloy are included in the models. Material properties used in the seismic analyses were:

Density	=	409 PCF
Young's Modulus	=	13×10^6 PSI
Shear Modulus	=	5×10^6 PSI

Densities used in the analyses not mentioned above are:

H_2O	=	62 PCF
UO_2	=	643 PCF

4.2 Seismic Input

The dynamic analysis of the Phase II Spent Fuel Racks included considerations for lift-off and friction, which is a non-linear response situation. As such, a time-history analysis in addition to a response-spectral analysis was required. This necessitated the generation of appropriate time-histories of accelerations at floor elevation 298 feet, for use as input motions in the Spent Fuel Rack analyses.

Summarized below is the sequence of activities followed to generate these time-histories:

(a) Submit the Reactor Building Mathematical Model to time-histories of accelerations representing the Site Base Input Seismic Motion.

(b) Extract from the analysis in (a) the time-history of accelerations at the spent fuel pool Floor Elevation 298 feet.

(c) Generate a Floor Response Spectrum (for the appropriate Single Degree of Freedom Oscillator Damping Ratio) from the time-history of accelerations at the spent fuel pool Floor.

(d) Broaden the Floor Response Spectra 15 percent per the recommendations in Regulatory Guide 1.122.

(e) Generate a Baseline Corrected Time-History of Acceleration whose Amplified Response Spectra (for the Single Degree of Freedom Oscillator Damping Ratio used above) envelopes the Broadened Design Response Spectra of step (d) above.

This procedure was carried out for both the North-South and East-West directions. The accelerations applied to the mathematical model and the damping ratios utilized conform to the Safe Shutdown Earthquake (SSE) condition. Per Regulatory Guide 1.61, the damping ratio during a Safe Shutdown Earthquake for the Spent Fuel Rack is 4 percent, the fundamental damping ratio used in this analysis for generation of response spectrum curves.

Building Models

In this analysis the Reactor Building, Reactor Pressure Vessel and shield wall models were coupled and analyzed as one complete model. Thus, the resulting response spectra included the dynamic interaction between the Reactor Building and the Reactor Pressure Vessel related structures.

Conservatism was incorporated in the analysis due to the utilization of 5 percent building damping (as opposed to the 7 percent building damping as listed in Regulatory Guide 1.61) in generating time-history of accelerations at the spent fuel pool Floor.

4.2 Seismic Input (Continued)

Modeling with a lower damping ratio inherently results in motions of higher amplitudes.

Base Input Records

The Base Input Records were time-histories of accelerations representing synthetic earthquakes. For conservatism and to be in accordance with the Standard Review Plan and latest NRC recommendations, the synthetic earthquake records for Nine Mile Point Nuclear Station Unit 2 were used. These records envelope the Nine Mile Point Nuclear Station Unit 1 Site Design Response Spectra and the Regulatory Guide 1.60 spectral curves, and have a base acceleration of 0.15g. To decrease the over conservatisms incorporated in the analysis by the use of 0.15g base acceleration (as opposed to the Nine Mile Point Nuclear Station Unit 1 FSAR base acceleration of 0.11g) a factor of 0.73 (the ratio of 0.11g over 0.15g) was applied when utilizing these acceleration records.

4.3 Seismic Loads

Seismic analysis of the fuel racks was performed using both the equivalent static response spectra method and the detailed non-linear time history approach.

Vertical Seismic

The vertical natural frequency is high enough to justify analysis using an equivalent static response treatment instead of a more detailed modal or time-history analysis treatment as per Standard Review Plan, Section 3.7.2. In addition, the vertical seismic accelerations are demonstrated to be less than 1 g, and the fuel bundles will remain in contact with the rack bottom plates for this event. Vertical reaction loads derived from this seismic acceleration are combined, using the square root sum of the squares method with the horizontal seismic loads.

Horizontal Seismic Analysis

This analysis used the time history method of analysis because of the non-linearities inherit in the Spent Fuel Storage Racks. These include:

1) Fuel-to-rack wall impacts;

2) Rack sliding;

3) Gaps between the racks and lateral seismic restraints;

4) Vertical impact due to rack tipping.

4.3 Seismic Loads (Continued)

A special purpose computer program (RACKO) developed specifically to analyze fuel storage rack behavior resulting from seismic disturbances, was used in this analysis. This program solves the equations of motion explicitly using Euler's extrapolation formula.

The fuel assembly was considered to rest in the rack cell at its center with a simple support (hinge) between the two and a gap between the fuel assembly and the rack cell wall along its length. With the seismic disturbance, the gaps between the fuel assembly and the cell walls can close, causing impact. However, the space between the fuel assembly and the cell wall is filled with water allowing the fuel assembly and the cell wall to move relative to each other. Hydrodynamic forces are set up due to the acceleration of the water. These forces are exerted on the fuel assembly and rack structure, which tend to mitigate impact forces.

Since the racks are confined within a rectangular pool, hydrodynamic forces are also generated between the rack and the pool walls. Sliding between the rack and the pool floor, and rack overturning were also considered.

4.4 Analytical Model

Lumped mass models were used to represent the rack with 216 fuel positions in the North-South and East-West directions. Differences in the models are in the magnitudes of the beam stiffnesses and support conditions. The rack and the fuel assemblies are represented as beams connecting masses (with all of the fuel assemblies in the rack lumped into a single beam). The rack and fuel assembly nodes can come into contact by means of compression-only springs with initial gaps. The gaps were determined to be 0.30 inches for unchanneled fuel and 0.22 inches for channeled fuel.

Springs represented the pedestals loaded initially by the submerged weight of the rack and are only capable of being loaded in compression. Using a mass representing a vertical degree of freedom, the effects of rack tilt were addressed.

In the North-South direction, the southern spring represented the stiffness of the seismic restraints including a 0.25 inch gap to allow for thermal expansion. Shims will be installed against the wall to ensure a 0.25 inch gap. The northern spring represented the stiffness of the seismic adapters and seismic beams. These will be installed without a gap.

The East-West direction has a 0.25 inch gap on each side of the rack. The spring stiffnesses included the stiffness of the rack bottoms. These stiffnesses were varied to reflect the position of the rack in the pool. These assumptions created additional conservatisms in the analysis because the cumulative effects of the gaps were not considered.

4.4 Analytical Model (Continued)

Damping values used for this analysis were taken from Regulatory Guide 1.61. Considering the rack to be a welded structure, the percent of critical damping used was 2 percent for Operating Basis Earthquake and 4 percent for Safe Shutdown Earthquake.

Friction at the rack base was represented by a special friction element. The normal force on this element is the force in the pedestal springs, which can be greater than the deadweight due to rack tilt. The friction factor was varied from 0.2 to 0.8 to bracket the most severe loadings. The low factor of 0.2 resulted in the largest loads on the lateral seismic restraints. Conversely a large friction factor resulted in the largest friction forces carried by the pedestals. In performing these runs, it was determined that a friction factor of 0.4 to 0.5 was sufficient to keep the racks from sliding. For cases above 0.5, the friction elements were replaced by a spring representing the horizontal stiffness of the pedestals. Hydrodynamic masses were included to represent the coupling between rack-fuel and rack-wall elements.

Seismic input consisted of the pool floor accelerations as a function of time for both the North-South and East-West Safe Shutdown Earthquake conditions (discussed in Section 4.2).

4.5 Analysis Results

Horizontal seismic analysis was performed for fuel stored both with and without channels on the rack with 216 fuel positions . Results indicated that because the channeled fuel storage configuration has small gaps containing water between the assembly and the rack wall, fluid coupling becomes significant and tends to reduce the impact loading. Fuel stored as unchanneled bundles did not see this effect. Thus, it generated higher rack loads even though the unchanneled Fuel Assemblies had a smaller mass. The results for the rack with 216 fuel positions conservatively were used for the rack with 198 fuel positions, since there is no basic structural difference between the two, and the mass of the rack with 198 fuel positions is about 90 percent of the rack with 216 positions.

The channeled fuel case, which is characterized by a higher moment of inertia for each Fuel Assembly and a much higher fluid coupling between each Fuel Assembly and the rack walls resulted in much smaller reaction loads, even though the mass of the channeled Fuel Assembly is about 10 percent greater than for the unchanneled Fuel Assembly.

5.0 POOL STRUCTURAL ANALYSIS

A three dimensional finite element model (see Figures 10, 11 & 12) and structural analysis were performed for the spent fuel pool and primary support columns. The objective of the analysis was to provide a structural assessment of the increased load resulting from the installation of high density spent fuel storage racks in the south half of the pool. The analysis concentrated on the spent fuel pool floor beams, slab panels and the primary columns under the pool floor. In addition, the stainless steel liner and pool wall base section were considered.

The internal stress (load) results and the load combinations for each load case were reviewed for each element in the three dimensional spent fuel pool finite element model.

5.1 Material Properties

The spent fuel pool and support columns are constructed of reinforced concrete. The spent fuel pool has a 1/4 inch stainless steel liner plate.

Concrete:

Ultimate strength (f'c) = 3500 PSI
Young's Modulus (Ec) = 3.41×10^6 PSI
Poisson's Ratio (vc) = 0.2
Density (1.05 x 145) = 152 PCF
(factor of 1.05 accounts for rebar and liner)

Reinforcing Steel (ASTM A15 Grade 40):

Yield strength (fy) = 43,000 PSI
(minimum per mill test reports)

The 1/4 inch liner plate is Type 304 Stainless Steel.

5.2 Parameters

The finite element model of the spent fuel pool used three dimensional shell elements to simulate the concrete floor and walls. Three dimensionsal beam elements were used to simulate the concrete columns.

The ultimate strength design method was used to evaluate the factored and service loads. Rules for the development of the cross-sectional strengths of the reinforced concrete slabs, beams, walls and columns were taken from ACI 318-77 and ACI 349-76.

5.2 Parameters (Continued)

The ACI Codes permit the use of a linear elastic model to simulate a reinforced concrete structure. Reinforcement is not simulated directly, nor is any tensile cracking in the concrete. However, both reinforcement and cracking are considered in the Code rules used to develop the cross-sectional strengths. Thus, the finite element model is a vehicle for obtaining a three dimensional internal load distribution that satisfies global equilibrium and geometric compatibility per Section 13.3.1 of ACI 318-77.

The primary support columns of the spent fuel pool were simulated. Column bases were assumed to be supported in translation (horizontal and vertical) but not in rotation nor torsion (twist). Column interfaces at floor elevations between the base and the pool were assumed to be supported horizontally but free vertically.

The southwestern half of the spent fuel pool floor and the southern half of the spent fuel pool west wall are integral with the massive biological shield wall. This interface was assumed to be a major support fixed in horizontal and vertical translation (but not in rotation). Vertical support for the spent fuel pool floor from the wall under the northern half of the west wall was included. The spent fuel pool walls were assumed to be supported horizontally at floor elevation interfaces with the reactor building.

The loads considered were: dead loads due to the structural weight of the spent fuel pool and the existing and new racks and fuel; hydrostatic loads from the pool water level of 37 feet, 10 inches; live loads due to the Cask Drop Protection System, fully loaded cask and embedment loads on the underside of the spent fuel pool floor; seismic loads due to Operating Basis Earthquake and Safe Shutdown Earthquake; thermal loads due to normal operation and loss of cooling accident and cask drop accident loads. The Spent Fuel Racks, Fuel, Assemblies and the Cask Drop Protection System were classified as permanent equipment, hence, considered dead load. Each of these loads were considered and the results combined using the combinations and load factors shown below.

Rack seimsic loads were obtained from time history studies which considered non-linear effects such as tipping, rattle, fluid coupling and friction (rack sliding). Maximum peak values were conservatively applied statically to the spent fuel pool model. Since the method of rack and Cask Drop Protection System seismic analyses are dependent on location and direction, seismic loads were run in each direction (vertical, north, south, east and west) These seismic runs were then conservatively combined using the Square Root of the Sum of the Square method, taking the absolute largest component in the East-West or North-South directions.

5.2 Parameters (Continued)

Below are the controlling load combination cases considered in the evaluation. These limiting cases are:

CASE	DEFINITION
A-1	1.4D + 1.7L + 1.4F + 1.7E
A-2	1.4D + 1.7L + 1.4F - 1.7E
B-1	.75 (1.4D + 1.7L + 1.4F + 1.4To + 1.7E)
B-2	.75 (1.4D + 1.7L + 1.4F + 1.4To - 1.7E)
C-1	D + L + F + To + E'
C-2	D + L + F + To - E'
D-1	D + L + F + To + P-CENTER
D-2	D + L + F + To + P-EDGE
E-1	D + L + F + Ta + 1.15E
E-2	D + L + F + Ta + 1.15E

where
E = Operating Basis Earthquake loads
E' = Safe Shutdown Earthquake loads
To = normal operating thermal load
Ta = loss of cooling accident thermal load
D = dead load
L = live load
F = hydrostatic load
P = cask drop loads

The above load combinations are in accordance with ACI 349-76. These differ from those load combinations presented in the Standard Review Plan, Section 3.8.4 in the following ways:

1. Load combinations involving the abnormal thermal load, Ta, in combination with Safe Shutdown Earthquake, E', were eliminated. Justification for this is the low probability of the simultaneous occurrence of the two independent extreme accident conditions. The spent fuel pool Cooling System is redundant and single failure proof and the piping is designed to Seismic Category I requirements. Thus, an earthquake would not be expected to lead to failure of the spent fuel pool Cooling System.

2. The load factor on the Operating Basis Earthquake, E, is reduced from 1.9 to 1.7 and the load factor on the normal thermal load, To, is reduced from 1.7 to 1.4. Using the higher load factors presented in the Standard Review Plan, in combination with the conservatisms in the analysis, where they pertain to the application of the seismic and thermal loads, would have created conditions of implied overstress.

5.2 Parameters (Continued)

The structural evaluations for torsion, shear, flexure plus compression and flexure plus tension were performed for each major reinforced concrete component of the Spent Fuel Pool. These were specifically the floor beams, floor slabs and primary support columns. The minimum margins (worst case) for all sections have load to strength ratios less than or equal to 1.0 and meet the code limits. A plan view of the spent fuel pool floor showing the beams and slabs and an isometric view of the columns are shown in Figures 10, 11 and 12.

The 304 stainless steel liner was evaluated for loads induced by thermal conditions and global deformations of the spent fuel pool floor. The basic assumption for global deformations of the spent fuel pool floor was that the strain in the liner is equal to the strain in the concrete floor. Under combined thermal and global strain conditions, the maximum stress intensity in the liner is 71 percent of the allowable stress intensity. These self-limiting stresses, if water pressure and rack presence are neglected, can cause the liner to buckle with a maximum vertical displacement of 0.417 inches. Since there is a clearance of 12 inches between the underside of the racks and the liner, the buckled liner will not alter the local cooling flow and will not affect the functional integrity of the liner.

Tensile and shear stresses in the fusion welds attaching the liner to the test channels are well within code allowables. The maximum pullout load on the anchor bolts attaching the liner and test channels to the concrete floor, under maximum design conditions, is 77 percent of the yield strength. Local stresses in the liner due to the rack reactions are addressed in Section 2.5.

The spent fuel pool walls are not significantly loaded by the racks. Very conservative calculations that bracket the shear and flexure on the wall base section, show that the spent fuel pool walls have sufficient strength to carry the "worst" combination of rack contact seismic and fluid loads.

5.3 Conclusion

The spent fuel pool walls, slabs and floor beams meet the applicable requirements and limits of Codes of ACI-349-76 and ACI-318-77 for torsion, shear, tension, compression and flexure for the specified service and factored load cases. The pool liner meets the limits within Appendix F, Section III of the ASME Code for Level D conditions.

The spent fuel pool columns meet ACI Code limits for bi-axial shear, axial compression and bi-axial flexure for the specified service and factored load cases. There are no load cases which result in tensile loadings in the columns. Torsional loading on the columns is negligible.

The stainless steel liner meets the allowable stress intensity for loads induced by thermal conditions and strain interactions with the concrete floor. The liner retention welds and anchor bolts also meet the allowables for tension and shear loads. In compression, the liner will buckle as indicated. However, liner integrity will remain.

The base section of the pool walls have sufficient strength to carry the "worst" combination of rack seismic reactions and fluid pressures.

6.0 THERMAL-HYDRAULIC ANALYSIS

The decay heat rates for the spent fuel pool were calculated for the normal refueling and emergency core offload conditions. An independent verification of the existing heat exchanger data was performed. Limits on the spent fuel pool bulk temperatures were calculated for the specified mass flow rates and design bases decay heat loads. The spent fuel pool heat-up rate and time until pool boiling following the loss of spent fuel pool cooling transient were also computed.

Two recirculation paths exist. The first is a local path involving the hottest spent fuel assembly and an adjacent long-term cooled spent fuel assembly. The second is a more complex path with under-the-rack flow. As the coolant temperature increased, the maximum cladding temperatures were evaluated for the hottest fuel assemblies for normal refueling and full core offload conditions.

Gamma heating of the poison cell and intercellular water adjacent to the hot fuel assembly were considered. The poison cell flow holes were sized according to the coolant flow rate and the enthalpy rise in the hottest spent fuel assembly. Temperature profiles in the fuel cell and poison cell walls, fuel assembly, water gaps, and Boraflex poison inserts were then found.

6.1 Scope

The scope of this analysis covered the following:

a. Computation of decay heat loads for the spent fuel pool in accordance with the NRC's Position Standard ASB 9-2.

b. Verification of the existing heat exchanger performance.

c. Determination of pool bulk temperatures for normal 200 bundle refueling and 532 bundle off-core loading decay heat loads and cooling conditions.

d. Temperature changes and heat-up rates for the loss of spent fuel pool cooling transient under normal refueling and full core offload conditions.

e. Recirculation flow characteristics in the hottest and average spent fuel assemblies to determine local coolant temperature changes and peak cladding temperatures. Also examined were local path and under-rack path flow characteristics.

f. Investigation of gamma heating in the poison cell adjacent to the hot assembly to size the inlet orifice hole(s) for the poison cell.

g. Convective and conductive cooling of intercellular water gaps.

h. Determination of the temperature distributions in the fuel cell, poison cell walls and in the poison insert materials.

The methods used for analyzing the thermal and hydraulic aspects of the spent fuel pool involve relatively uncomplicated correlations for friction factors, loss coefficients, and heat transfer coefficients that make a detailed computer analysis unnecessary.

6.2 Design Criteria

The design criteria used for the thermal and hydraulic analyses of the spent fuel pool are in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", issued April 14, 1978.

The following are the design bases for the Thermal-Hydraulic Analysis:

Decay heat loads for a full pool were determined in accordance with the NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling", Section 9.2.5-8a of the Standard Review Plan. Full pool decay heat loads and temperatures were computed for the following cases:

1) A normal refueling discharge of 200 fuel assemblies starting ten days after reactor shutdown. The remainder of the pool is filled with normal refueling discharges cooled 24 months or multiples thereof. The pool bulk outlet temperature is limited to 125°F. The worst case occurs in 1994 using projected fuel cycle plans.

2) An equilibrium core of 532 fuel assemblies is discharged beginning 10 days after reactor shutdown during the last refueling in 1994. The pool is completely filled with 2,776 fuel assemblies. The pool bulk outlet temperature is limited to 125°F with two heat exchangers running or 159°F with one heat exchanger in operation.

6.2 Design Criteria (Continued)

Additional assumptions used for the thermal and hydraulic analysis of the spent fuel pool are as follows:

a. In determining the pool bulk temperatures, only one (of two) cooling loops were assumed to be operational.

b. The thermal inertias of the concrete walls and the coolant and piping outside the pool boundaries are neglected in the transient heat-up analysis.

c. The pool surface was not assumed to mix to a lower pool bulk temperature in the heat-up analysis following the loss of spent fuel pool cooling transient.

d. Decay energy was assumed to be absorbed in the fuel and surrounding coolant for the hot assembly or natural circulation analysis (In reality, some gamma radiation will be absorbed in the adjacent poison cell).

e. The gamma decay heat absorbed in the poison cell is proportional to the mass densities of the materials in the Spent Fuel Pool. (In reality, most of the gamma radiation never leaves the fuel cell due to strong uranium attenuation). Gamma heating proportional to the mass fraction is roughly equivalent to the assumption of uniform gamma flux in the repeating unit cell.

The major concerns in the thermal and hydraulic analysis were verifying that the cladding and coolant temperatures do not become high enough to cause boiling, and that the void fraction is small enough to be neglected in nuclear criticality calculations. In the event of the loss of spent fuel pool cooling transient, the heat-up rate must be slow enough to allow an alternate coolant system to be connected and operating before pool boiling occurs.

6.3 Conclusion

The detailed thermal and hydraulic analyses described addressed the concerns, intent and design bases of the NRC's Position Paper "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

Based on these analyses, the spent fuel pool can be adequately cooled in accordance with the suggested regulatory standards of the Nuclear Regulatory Commission.

7.0 RADIOLOGICAL CONSEQUENCES

The principle source of radiation levels observed at the surface of the spent fuel pool is generally due to radionuclides in the pool water. The observed dose rate has been typically less than 5 mR/hr. The radionuclides are removed from the water by the spent fuel pool demineralizer with the need for changing the filter resin determined by the pressure drop across the filter. Increased fuel storage may result in an increased frequency of changing the filter resin but is not expected to result in any increase in the radionuclide concentrations or in subsequent radiation levels at the surface of the water.

The top of the fuel assemblies stored in the spent fuel storage racks are at least 25 feet below the surface of the water which is the same as the original. The 25 foot water shield reduces the direct radiation from the stored fuel assemblies to values which are negligible when compared to background.

In the Phase I racks, the sides of the fuel bundles stored closest to the walls were approximately 11 inches from the inside concrete surface of the pool. The Phase II fuel racks will reduce this distance to 4 inches. The reduction in distance and the closer fuel bundle spacing should result in no measurable increase in radiation levels outside the spent fuel pool. Attenuation factor for the wall of the pool is 10^8. The attenuation factor of 11 inches of water is 4. This factor becomes less than 2 for 4 inches of water. This change is estimated to be below measurement accuracy. This will be verified as fuel is placed close to the wall.

Therefore, for the reasons mentioned, the increased fuel storage will have essentially no impact on the radiation levels at the surface of the water or outside the spent fuel pool walls.

8.0 NRC QUESTIONS

The questions herein listed were raised by Mr. P. Polk, past Nuclear Regulatory Commission Project Manager for Nine Mile Point Unit 1, during a telephone conference with members of the Niagara Mohawk Power Corporation staff on April 7, 1982. At a subsequent meeting on April 13, 1983 with Mr. R. Hermann, present Nuclear Regulatory Commission Project Manager for Nine Mile Point Unit 1, requested responses to the questions listed below.

Question 1

Have the liquid waste treatment systems changed since July, 1974?

Response

The liquid waste treatment system has been upgraded since July 1974. Dow and Stock Systems have been installed to treat and solidify wastes faster and more effectively. Additionally, engineering is underway to increase liquid radwaste storage capacity. The storage of additional spent fuel does not represent a significant increase in waste volume.

Question 2

Have or are the thermal stresses been included in the analysis of the racks in the pool?

Response

Thermal effects have been included in the analysis of the racks under an assumed pool water temperature of 200°F.

Question 3

a. Has a liner crack analysis been performed and;

b. What is the process whereby spent fuel pool leaks are detected?

Response

a. The liner has been analyzed for stresses and possible rupture. Results indicate that liner integrity will be maintained and overstressing does not occur as indicated in Section 5.0 of the Submittal.

b. Each liner seam on the pool floor and walls contain leak-off channels. These channels connect to open telltale drains where leaks are detected by visual inspection. A level control valve fed by the Condensate Transfer System, maintains pool water elevation.

Question 4

Show analyses of liner and liner support stresses.

Response

The liner and liner supports which are anchored into the concrete pool floor have been analyzed for loads imposed through swing bolt brackets, frictional forces, global floor deformation and thermal conditions as described in Sections 2.0 and 5.0 of the submittal.

Question 5

Regarding the thermal hydraulic analysis, show the maximum pool temperature and the method and the time it would take for that maximum temperature to be reached?

Response

The normal bulk fuel pool temperature, prior to any refueling, is 101°F, with one heat exchanger operating.

With full core discharge in 1994, wherein the pool would contain 2,776 assemblies, the bulk pool water temperature would be 125°F with two heat exchangers operational. With the loss of the heat exchanger, the bulk pool temperature would rise from 125°F to 159°F in approximately 26 hours.

Question 6

Please provide a draft of the structural analysis of the pool.

Response

The structural analysis is described in Section 5.0 of the Supplemental Submittal and in the attached Structural Reports.

Question 7

Will neutron attenuation tests be performed at the site?

Response

Our letter of December 20, 1978 indicated that neutron attenuation tests would be performed on 10 percent of the storage locations at the manufacturers plant. In addition, it was indicated that site verification would be performed on at least 5 storage locations in each rack module. Our current plans call for neutron attenuation tests to be performed at the site on 15 percent of the fuel assembly positions within each rack module.

Question 8

What will be the poison surveillance procedure and method?

Response

Our letter dated February 26, 1981 from D. P. Dise to T. A. Ippolito of the Nuclear Regulatory Commission indicated that poison surveillance will not be conducted at Nine Mile Point Unit 1.

Question 9

What is the comparative design between the Phase I north half of the pool and the Phase II south half design?

Response

Refer to Section 1.0 - (SUMMARY) of this submittal.

Question 10

Is rack rotation or tipping being considered in the analysis?

Response

Rack rotation and resultant impacting of the racks is included in the analysis. The analysis is described in Section 2.0 of this Supplemental.

9.0 INSTALLATION

A potential fuel accident will not occur from the installation of racks and working platforms in the south half of the pool because all fuel will be stored in the north half and handling procedures will prevent equipment travel over racks containing fuel.

The removal of 26 original General Electric spent fuel racks will clear the south half of the pool in preparation for the work. The Phase I racks are and will continue to be seismically supported and independent of the Phase II racks. There will be no mechanical interaction between non-poison and poison racks as indicated in Figure 1.

Two lugs will be welded to the pool liner to provide lateral seismic restraint for two of the eight racks as indicated in Figure 2. A liner repair procedure will be available in the event that welding to the liner would cause leakage.

Shims, seismic beams, brackets, support legs and wall restraints will be pre-placed in the pool prior to installing the racks and working platforms as indicated in Figures 2 through 5. The racks will be traversed from the south west side of the pool and placed in their locations with appropriate gaps.

A step-by-step installation procedure has been written to ensure safety and minimize radiation exposures.