Attachment I to IPN-88-018

Proposed Technical Specifications
Spent Fuel Pool Storage Capacity Expansion

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
DPR-64

LIST OF FIGURES

LIST OF FIGURES

8. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

9. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 145 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 365 hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal absorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications. In the event that more than one region of fuel (72 assemblies) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before the interval of 267 hours has elapsed after shutdown.

10. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of the reactor pressure vessel flange.

11. Hoists or cranes utilized in handling irradiated fuel shall be dead-load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test and prior to fuel handling. A test of interlocks and overload cutoff devices on the manipulator shall also be performed.

12. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

13. To ensure redundant decay heat removal capability, at least two of the following requirements shall be met:

a. No. 31 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

b. No. 32 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

c. The water level in the refueling cavity above the top of the reactor vessel flange is equal to or greater than 23 feet.

B. If any of the specified limiting conditions for refueling are not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

C. During fuel handling and storage operations, the following conditions shall be met:

1. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used.

2. The spent fuel cask shall not be moved over any region of the spent fuel pit which contains irradiated fuel. Additionally, if the spent fuel pit contains irradiated fuel, no loads in excess of 2,000 pounds shall be moved over any region of the spent fuel pit. This prohibition does not apply to the movement of the existing high density or replacement maximum density spent fuel storage racks over the spent fuel pit during the storage rack replacement effort, provided that the fuel stored in the spent fuel pit has been subcritical for a minimum of 120 days.

3. During periods of spent fuel cask or fuel storage building cask crane movement over the spent fuel pit, or during periods of spent fuel movement in the spent fuel pit when the pit contains irradiated fuel, the pit shall be filled with borated water at a concentration of $>$ 1000 ppm.

4. Whenever movement of irradiated fuel in the spent fuel pit is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated in the storage rack.

5. Hoists or cranes utilized in handling irradiated fuel shall be deadload tested before fuel movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the fuel handling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test prior to fuel handling.

6. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the periods of inoperability.

7. <u>Fuel Storage in High Density Spent Fuel Racks Only:</u>

   Fuel assemblies to be stored in the spent fuel pit can be categorized as either Category 1, 2 or 3 based on burnup and initial enrichment as specified in Figure 3.8-1. Category 2 fuel shall be loaded into the spent fuel pool storage locations in a checkerboard fashion with the intermediate storage locations containing Category 1 fuel, non-fuel materials or left empty. Unless restricted by the above, Category 1 or 3 fuel can be stored in any location in the spent fuel pool.

8. <u>Fuel Storage in Maximum Density Spent Fuel Racks Only</u>:

   Fuel assemblies of initial enrichment less than or equal to 4.5 w/o U-235 can be stored in Region 1 (rows SS-ZZ, columns 35 - 64) of the spent fuel storage racks. Fuel assemblies to be stored in Region 2 (rows A-RR, columns 1-34) of the spent fuel storage racks shall have a minimum burnup exposure as a function of initial enrichment as specified in Figure 3.8-2. The locations of Regions 1 and 2 of the spent fuel storage racks are shown in Figure 3.8-3.

<u>Basis</u>

The equipment and general procedures to be utilized during refueling, fuel handling, and storage are discussed in the FSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling, fuel handling, reactor maintenance or storage operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

The shutdown margin indicated will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling the reactor refueling cavity is filled with approximately 342,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. A shutdown margin of 10% $\Delta K/K$ in the cold condition with all rods inserted will also maintain the core subcritical even if no control rods were inserted into the reactor.(2) Periodic checks of refueling water boron concentration and residual heat removal pump operation insure the proper shutdown margin. The requirement for direct communications allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 145-hour decay time following the subcritical condition and the 23 feet of water above the top of the reactor pressure vessel flange bounds the assumptions used in the dose calculation for the fuel-handling accident. The 145 hour decay time is based on limiting calculated worst - case spent fuel pool temperature rise to 150°F during normal refueling conditions.

The waiting time of 267 hours required following plant shutdown before unloading more than one region of fuel from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR. The calculations confirming this are based on an inlet river temperature of 87.8°F, service water flow to the component cooling heat exchangers of 7000 gpm (FSAR) and component cooling flow to the Spent Fuel Pit heat exchanger of 2800 gpm (FSAR).

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite dose to within acceptable limits in the event of a fuel-handling accident. The fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved or whenever the replacement or existing fuel storage racks are moved over the spent fuel pool. However, if the irradiated fuel has had a continuous 45 day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. Fuel Storage Building isolation is actuated upon receipt of a signal from the area high activity alarm or by manual operation. The emergency ventilation bypass assembly is manually isolated, using manual isolation devices, prior to movement of any irradiated fuel. This ensures that all air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers. The ventilation system is tested prior to all fuel handling activities to ensure the proper operation of the filteration system.

When fuel in the reactor is moved before the reactor has been subcritical for at least 365 hours, the limitations on the containment vent and purge system ensures that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere.

The limit to have at least two means of decay heat removal operable ensures that a single failure of the operating RHR System will not result in a total loss of decay heat removal capability. With the reactor head removed and 23 feet of water above the vessel flange, a large heat sink is available for core cooling. Thus, in the event of a single component failure, adequate time is provided to initiate diverse methods to cool the core.

The minimum spent fuel pit boron concentration and the restriction of the movement of the spent fuel cask over irradiated fuel were specified in order to minimize the consequences of an unlikely sideways cask drop.

Fuel assemblies whose initial enrichment is greater than 3.5 w/o high density U-235 but less than or equal to 4.3 w/o can be stored in the existing spent fuel storage racks providing they are placed in a checkerboard array with fuel whose initial enrichment and burnup are sufficient to ensure that K_{eff} is less than 0.95 with no soluble boron present. This is ensured by categorizing the fuel whose initial enrichment is greater than 3.5 w/o U-235 but less than or equal to 4.3 w/o and whose burnup is below the curve of Figure 3.8-1 as Category 2. This fuel can be stored by checkerboarding with Category 1 fuel which is defined as fuel whose initial enrichment and burnup, place it on or above and to the left of the curve in Figure 3.8-1. Category 3 fuel which is less than or equal to 3.5 w/o U-235 and below the curve of Figure 3.8-1 cannot be used in the checkerboard with Category 2 fuel. Any Category 1 or 3 fuel can continue to be stored on a repeating x-y array with other non-fuel material or empty locations can be utilized in place of Category 1 fuel.

As shown in Figure 3.8-3 the maximum density spent fuel storage racks consist of two regions: Region 1 (rows SS - ZZ, columns 35-64) and Region 2 (rows A-RR, columns 1-34). Fuel assemblies of initial enrichment of less than or equal to 4.5 w/o U-235 may be stored in Region 1 of the replacment maximum density spent fuel storage racks. Fuel assemblies to be stored in Region 2 of the replacment racks must have a minimum burnup exposure as a function of initial enrichment as specified in Figure 3.8-2. Administrative controls will provide verification that each fuel assembly to be placed in Region 2 of the replacement racks satsifies the burnup criterion.

When the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops incorporated in the bridge rails make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit. Thus, it will be possible to handle the spent fuel cask with the 40-ton hook and to move new fuel to the new fuel elevator with a 5-ton hook, but it will be impossible to carry any object over the spent fuel storage area with either the 40 or 5-ton hook of the fuel storage building crane.

During the spent fuel storage rack replacement effort, the mechanical stops incorporated in the crane bridge rails will be removed. However, administrative controls will preclude the movement of the existing high density or replacement maximum density racks directly over spent fuel assemblies.

Dead load test and visual inspection of the hoists and cranes before handling irradiated fuel provide assurance that the hoists or cranes are capable of proper operation.

<u>References</u>

(1) FSAR - Section 9.5.2
(2) FSAR - Table 3.2.1-1

5.3 REACTOR

Applicability

Applies to the reactor core, and reactor coolant system.

Objective

To define those design features which are essential in providing for safe systems operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.(1)

2. The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235.(2)

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 4.5 weight percent of U-235.

4. Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16 and 20-rod clusters, which are located in vacant rod cluster control guide tubes.(3) The burnable poison rods consist of borosilicate glass clad with stainless steel.(4) Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control.

5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specification

1. The spent fuel pit structure is designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel pit has a stainless steel liner to insure against loss of water.

2. The spent fuel storage racks are designed to assure $K_{eff} < 0.95$ if the assemblies are inserted in accordance with Technical Specification 3.8. The capacity of the spent fuel pit is 1345 assemblies with the maximum density storage racks installed. The new fuel storage racks are designed to assure $K_{eff} < 0.95$ and their capacity is 72 assemblies.

3. Whenever there is fuel in the pit (except in the initial core loading), the spent fuel storage is filled and borated to the concentration to match that used in the reactor cavity and refueling canal during refueling operations.

4. Fuel assemblies that contain more than 57.2 grams of uranium - 235, or equivalent, per axial centimeter of fuel assembly shall not be stored in the spent fuel pit.

Attachment II to IPN-88-018
Safety Evaluation of Proposed Changes to Technical Specifications Related to Spent Fuel Pool Storage Capacity Expansion

Enclosure 1: Safety Analysis Report

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
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I. Description of Change

This application seeks to amend Sections 3.8, 5.3 and 5.4 of the Operating License to allow for the replacement of the existing high density spent fuel storage racks with maximum density storage racks. This replacement will result in a spent fuel pool storage capacity increase from 840 assemblies to 1345 assemblies.

The existing high density storage racks are arranged in twelve modules. In seven of the twelve existing modules, the center-to-center spacing of the fuel cells is 12 inches in either direction; in five of the twelve modules, the center-to-center spacing of the fuel cells is 12 inches in one direction and 11.25 inches in the other direction. The existing storage cells are formed from a type 304 stainless steel sheet of 0.150 inch minimum thickness with borated stainless steel plates welded to the cell at specified locations.

The replacement storage rack design is a free-standing welded honeycomb array of stainless steel boxes which has no grid frame structure. The storage racks are arranged and categorized in two regions based on fuel assembly enrichment and burnup. The nominal pitch for region 1 storage cells is 10.76 inches. The nominal pitch for region 2 storage cells is 9.075 inches. All storage cells are bounded on four sides by boral poison sheets, except on the periphery of the pool rack array.

Each of the replacement maximum density racks will be supported and leveled on four screw pedestals which bear directly on the pool floor. These free-standing racks are free to move horizontally. However, with only a 0.2 friction factor, there is no wall impact even assuming five (5) OBE and one (1) SSE earthquake events all added up in the same direction. Additionally, there is no rack-to-rack impact since the maximum density racks are designed to be installed with essentially no gap between the racks. The strong hydrodynamic coupling between the racks causes the racks to move together even when a full and empty rack are adjacent to each other. The seismic analysis (Reference 1) shows that rack-to-rack impact will not occur through the full range of realistically expected gaps between installed racks.

The maximum density storage racks are designed for a fuel enrichment of up to 4.5 w/o U-235, which is slightly higher than the currently allowable maximum of 4.3 w/o U-235. This application also seeks to increase the maximum fuel enrichment allowed in the spent fuel pool and the reactor core from 4.3 w/o to 4.5 w/o U-235.

Enclosure 1 to this safety evaluation provides a more detailed description of the spent fuel storage facility modification.

II. Evaluation of Change

Enclosure 1 to this safety evaluation provides a detailed presentation of the safety analyses performed in support of the spent fuel storage rack replacement. The discussion that follows provides a summary of the analyses presented therein.

Nuclear Criticality Analysis

The established acceptance criterion for criticality is that the neutron multiplication factor in spent fuel pools shall be less than 0.95, including all uncertainties, under all conditions. As noted above, the spent fuel storage racks are arranged and categorized in 2 regions. Region 1 is designed for safe storage of new fuel and spent fuel of any burnup. Region 2 is designed for safe storage of fuel which has accumulated a specified minimum burnup based on initial enrichment. The criticality analysis addresses postulated accidents and shows that K_{eff} in both regions is less than 0.95.

The criticality analysis included the double contingency principle of ANSI 8.1 - 1983 which requires two unlikely independent, concurrent events to produce a criticality event. The spacing of fuel assemblies acceptable for storage ensures that a subcritical array of K_{eff} less than 0.95 is maintained, assuming unborated pool water.

Decay Heat and Bulk Pool Temperature Analyses

The additional heat load introduced by the increased storage capacity necessitated an analysis of the adequacy of the existing spent fuel pool cooling system. The time-dependent pool water temperatures resulting from a normal refueling discharge and a full core discharge were calculated. For each given fuel discharge scenario, decay heat loads were calculated using NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling" (Reference 2). Per Standard Review Plan 9.1.3, "Spent Fuel Pool Cooling and Cleanup System" (Reference 3), a single active failure should be considered during the normal refueling discharge case only.

For the normal refueling discharge case, 76 assemblies with a burnup of 1050 days were discharged into the spent fuel pool, after decaying 145 hours in the reactor vessel. Subsequent to discharge, the spent fuel pool contained the 76 newly discharged assemblies, 1076 assemblies discharged after previous fuel cycles, and 193 empty cells representing the full core discharge capacity.

The calculated maximum pool bulk temperature resulting from this discharge scenario was 150°F, under the following assumptions: The 76 spent fuel assemblies are discharged instantaneously after the 145 hour decay time; the heat removal effectiveness of the spent fuel pool heat exchanger is 90% of design; no credit for heat loss to the pool walls and pool floor slab and by pool water evaporation; an initial spent fuel pool temperature of 100°F. Included in these assumptions is that only one of the two spent fuel pit pumps is operating, which is consistent with the Standard Review Plan 9.1.3 single active failure requirement. Assuming a complete loss of pool cooling commencing at the time of maximum pool bulk temperature, the pool temperature will increase at a rate of 7.30°F/Hr. For this case, 8.5 hours are available to re-establish pool cooling before bulk boiling occurs.

For the full core discharge case, 193 assemblies with burnups ranging from 666 days to 1050 days, were discharged into the spent fuel pool, after decaying 267 hours in the reactor vessel. Prior to discharge, all but the 193 empty cells contained fuel assemblies discharged after previous fuel cycles. The maximum pool bulk temperature resulting from this discharge scenario was 200°F, under the aforementioned assumptions. Assuming a complete loss of pool cooling commencing at the time of maximum pool bulk temperature, the pool temperature will increase at a rate of 14.6°F/Hr. For this case, 49.2 minutes are available to re-establish pool cooling before bulk boiling occurs.

The primary source of makeup water to the spent fuel pool is the Primary Water Storage Tank. Additional makeup water may be provided from the Refueling Water Storage tank or the city water supply.

It should be noted that the major component of the heat load to the pool is from the newly discharged assemblies. The contribution to the total heat load from the previously discharged assemblies is relatively insignificant due to the length of time that has elapsed since discharge. As such, the effect of the increased spent fuel pool storage capacity on bulk pool temperature analyses is not significant.

An evaluation indicated that this pool bulk temperature transient will not significantly affect concrete material properties. Following the postulated increase in pool bulk temperature, the pool boundary will maintain its structural integrity under the limiting load combination (ie safe shutdown earthquake and pool bulk temperature of 212°F).

Localized Thermal-Hydraulic Analyses For Spent Fuel Pool

The primary purpose of the localized thermal-hydraulic analysis is to determine the maximum spent fuel clad temperature. The spent fuel storage racks must allow adequate cooling by natural circulation and by forced spent fuel cooling system flow. The coolant should remain subcooled at all points within the pool.

The methods used for analyzing the localized thermal-hydraulic aspects of the spent fuel pool involve relatively uncomplicated correlations for friction factors, loss coefficients, and heat transfer coefficients. In this analysis, two recirculation paths are identified for the natural circulation cooling of the Indian Point 3 spent fuel assemblies. A local path where coolant is convectively driven up the hottest assembly and down a "cold" assembly is studied first. A second path flowing under the spent fuel racks, up the hot assemblies, into the mixing region above the racks, and finally down the South wall of the pool to complete the path was then modeled and analyzed. For the local path, the fuel assembly inlet temperature was taken to be the hottest pool bulk temperature of 200°F for full core unload condition. For the second path, the inlet temperature is taken to be the average for the pool. Apart from the estimation of the coolant inlet temperatures to the hot batch of spent fuel, these flow paths are decoupled from the cooling loop and spent fuel pool heat exchanger.

For the local path, the peak clad temperature for the full core discharge case was 237.2°F. The peak coolant temperature was 222.9°F. For the second path, the peak clad temperature for the full core discharge case was 222.9°F. The peak coolant temperature was 210.1°F. For both pathways, the peak temperatures are below the saturation temperature. As such, the coolant will remain subcooled at all locations.

<u>Fuel Handling Analysis</u>

The increased spent fuel pool capacity will not adversely affect the Fuel Handling Analysis presented in the FSAR. The design of the fuel handling crane is such that only one assembly can be handled at a time. The FSAR provides evaluations of the effect of dropping an assembly on another fuel assembly and on a rigid surface.

For the case where one fuel assembly is assumed to be dropped on top of another assembly, the impact load is transmitted through the nozzle and the reactivity control cluster (RCC) guide tubes of the struck assembly before any of the loads reach the fuel rods. As a result, a significant amount of kinetic energy is absorbed by the top nozzle of the struck assembly and bottom nozzle of the falling assembly, thereby limiting the energy available for fuel rod deformation. The results of this analysis indicated that the buckling load on the fuel rods was below the critical buckling load and stresses in the cladding were below yield. The analysis of a dropped fuel assembly striking a rigid surface considered the stresses in the fuel cladding and any possible buckling of the fuel rods between the grid supports. The results showed that the buckling load at the bottom section of the fuel rod, which would receive the highest loading, was below the critical buckling load and the stresses were below the yield stress.

For the purposes of evaluating the environmental consequences of a fuel handling incident, the FSAR assumed a conservative upper limit by considering the rupture of one complete fuel assembly. The calculated resultant dose would not exceed the 10 CFR 100 limits. The increased spent fuel pool capacity will not adversely affect the assumptions or the conclusions of this evaluation.

The increased spent fuel pool capacity will not affect the structural integrity of the individual fuel assemblies. As the aforementioned evaluations are based on the structural integrity of the fuel assemblies, the increased spent fuel pool capacity will not adversely affect these evaluations.

In the unlikely event of a fuel assembly drop on a storage rack, the resulting deformation will not result in the design criteria of K_{eff} 0.95 being exceeded. Three drop scenarios were postulated in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 4).

The first scenario involved a vertical drop of a fuel assembly on the top of the rack. The fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the fuel sufficient to preclude neutron coupling. Maximum expected deformation under seismic or accident conditions will not reduce the minimum spacing between the stored fuel assemblies. Consequently, fuel assembly drop accidents will not result in a significant increase in reactivity due to the separation distance. Furthermore, soluble boron in the pool water would substantially reduce the reactivity and assure that the true reactivity is always less than the limiting value for any conceivable dropped fuel accident.

The second scenario involved an inclined drop on the rack. As compared to the vertical drop scenario, the inclined drop will result in a much less severe impact force and rack damage.

The third scenario assumed a straight drop through an empty cell. This will result in high energy absorption in the cell bottom plate welds possibly leading to bottom plate weld failure. However, this drop will not result in a rack geometry deformation of a magnitude resulting in the criticality acceptance criterion being violated. The remaining energy is then absorbed by the pool floor without rupturing the liner.

Heavy Loads Evaluation

Technical Specification 3.8.C.2 provides that the spent fuel cask shall not be moved over any region of the spent fuel pit which contains irradiated fuel. Additionally, if the spent fuel pit contains irradiated fuel, no loads in excess of 2,000 pounds shall be moved over any region of the spent fuel pit. As the existing and replacement storage racks weigh in excess of 2,000 pounds an exception to this prohibition on movements of loads in excess of 2,000 pounds is being sought.

Sections 5.1.1, 5.1.2 and 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Reference 5), provide guidance for heavy load handling operations pursuant to a spent fuel storage rack replacement.

Section 5.1.2 provides four alternatives for assuring the safe handling of heavy loads during a fuel storage rack replacement. Alternative (1) of Section 5.1.2 provides that the control of heavy loads guidelines can be satisfied by establishing that the potential for a heavy load drop is extremely small as demonstrated by meeting the single-failure-proof crane guidelines. Alternative 1 is satisfied during the subject application.

NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants", provides guidance for design, fabrication, installation and testing of new cranes that are of a high reliability design. For operating plants, NUREG-0612, Appendix C, "Modification of Existing Cranes," provides guidelines on the implementation of NUREG-0554 for operating plants. An evaluation of storage rack movements by the fuel storage building crane for conformance with the NUREG-0612, Appendix C guidelines demonstrated that alternative (1) above is satisified i.e., the probability of a drop of a storage rack is extremely small. The fuel storage building crane has a rated capacity of 40 tons, which incorporates a factor of safety of five. The maximum weight of any existing or replacement storage rack is 17 tons. Therefore, the minimum safety factor is 11.8 for movements of the storage racks by the fuel storage building crane. This applies to non-redundant load-bearing components. Redundant special lifting devices, which have a rated capacity sufficient to maintain this minimum safety factor, will be utilized in the movements of the storage racks. As per NUREG-0612, Appendix B, this minimum safety factor ensures that the probability of a load drop is extremely low.

The existing mechanical stops will be removed so that the fuel storage building crane will have access to any location over the spent fuel pool. However, administrative controls, which incorporate predetermined safe load pathways, will ensure that at no time will any storage rack be moved directly over an irradiated fuel assembly.

Seismic Event Evaluation

A seismic analysis of the spent fuel storage racks was performed to determine the rack behavior and to ensure no loss of function resulting from a safe shutdown earthquake (SSE) of 0.15g horizontal and an operating basis earthquake (OBE) of 0.10g horizontal. The applicable vertical component was taken to be 2/3 of the horizontal component. A non-linear finite element computer program was used to analyze the horizontal disturbances, using time-histories synthesized from an equivalent static method using the peak response spectra. The seismic analysis determines the rack loads, sliding and lift-off in the three orthogonal directions. The loads are combined using the square root sum of the squares method. Sliding and lift-off results indicate that the racks will not impact the walls. Additionally, there is no rack-to-rack impact since the maximum density racks are designed to be installed with essentially no gap between the racks. The strong hydrodynamic coupling between the racks causes the racks to move together even when a full and empty rack are adjacent to each other. The seismic analysis (Reference 1) shows that rack-to-rack impact will not occur through the full range of realistically expected gaps between installed racks. The analysis also demonstrates that the racks will maintain their integrity during the postulated seismic events. The new racks are designed so that floor loading from the racks filled with spent fuel assemblies does not exceed the structural capacity of the spent fuel pool floor.

The maximum density spent fuel pool rack modules are designed to be free-standing, i.e., any single rack or combination of racks installed in the spent fuel pool is capable of withstanding a design basis seismic event without toppling or causing damage to fuel assemblies inserted within them. The existing spent fuel pool racks, on the other hand, are not free-standing but are instead provided with interties that provide the necessary support to prevent toppling or damage to fuel during a design basis seismic event. Furthermore, a previous analysis of the existing racks shows that four spent fuel pool rack modules in a square configuration, connected with interties, is sufficiently stable to be designated as free-standing.

The rack removal/insertion sequence will be designed with the aforementioned restrictions in mind. Specifically, no existing interties will be removed until the rack to which they are attached is ready to be lifted. Furthermore, the four-module configuration of the existing racks will be maintained as much as possible during the removal sequence. If, at any time during reracking, there exists a rack configuration less stable than the four-module configuration, it will be analyzed and any additional restraints required for ensuring seismic integrity will be provided as appropriate.

Fuel Enrichment Increase

The spent fuel pool criticality evaluations were performed assuming stored spent fuel with enrichments of up to 4.5 w/o U-235. If the spent fuel is stored in accordance with Figure 3.8-2, the design criteria of $K_{eff}<0.95$ will not be exceeded during normal operations and during design basis events.

The increased fuel enrichments of 4.5 w/o U-235 will not affect the maximum level of power operation. As such, the operating transient analyses, which are dependent on power level, are not impacted.

The slightly higher enrichments will facilitate extended fuel cycles. An extended fuel cycle will not increase the fuel rod gap activity since the activity reaches an equilibrium value prior to the end of the current fuel cycle. As such, the off-site dose consequences of a fuel handling accident will not be increased due to extended fuel cycle.

Section III - No Significant Hazards Evaluation

1. Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated.

Response:

The following potential accident scenarios have been identified as concerns that must be addressed in support of a proposed spent fuel pool storage capacity expansion.

1. Spent fuel assembly drop in the spent fuel pool.
2. Loss of spent fuel pool cooling system flow.
3. Seismic event.
4. Spent fuel cask drop.
5. Storage rack handling accident.

The probability of a spent fuel assembly drop in the spent fuel pool is a function of the structural integrity of the manipulator crane and the integrity of the crane-assembly coupling. The probability of such a drop is not affected by the rack design. Therefore, the rack replacement will not increase the probability of a fuel assembly drop. The consequences of a spent fuel assembly drop in the fuel pool are discussed in Section II and in Enclosure 1. The criticality acceptance criterion of $K_{eff} < 0.95$ was not violated for such a postulated drop. The radiological consequences of a fuel assembly drop are not changed from that provided in Indian Point 3 FSAR Chapter 14.2. Therefore, the rack replacement will not involve a significant increase in the consequences of a fuel assembly drop.

The probability of a loss of spent fuel pool cooling system flow is a function of the integrity of the cooling system itself. The probability of a loss of forced flow is not affected by the rack design. The consequences of a loss of spent fuel cooling system flow are discussed in Section II and Enclosure 1. As indicated therein there is sufficient time to provide an alternate means for cooling in the event of a failure in the cooling system. Furthermore, in the region of the fuel assemblies, the coolant will remain subcooled. Therefore, the rack replacement will not involve a significant increase in the consequences of a loss of cooling system flow accident.

The design of the storage racks has no effect on the probability of a seismic event occurring. A seismic analysis of the spent fuel storage racks was performed to determine the rack behavior and to ensure no loss of function resulting from a safe shutdown earthquake (SSE) of 0.15g horizontal and an operating basis earthquake (OBE) of 0.10g horizontal. The applicable vertical component was taken to be 2/3 of the horizontal component. A non-linear finite element computer program was used to analyze the horizontal disturbances, using time-histories synthesized from equivalent static method using the peak response spectra. The seismic analysis determines the rack loads, sliding and lift-off in the three orthogonal directions. The loads are combined using the square root sum of the squares method. Sliding and lift-off results indicate that the racks will not impact the walls. Additionally, there is no rack-to-rack impact since the maximum density racks are designed to be installed with essentially no gap between the racks. The very strong hydrodynamic coupling forces the racks to move together even when a full and empty rack are adjacent to each other. The seismic analysis (Reference 1) shows that rack-to-rack impact will not occur through the full range of realistically expected gaps between installed racks. The analysis also demonstrates that the racks will maintain their integrity during the postulated seismic events. The new racks are designed so that floor loading from the racks filled with spent fuel assemblies does not exceed the structural capacity of the spent fuel pool floor. Therefore, the rack replacement will not involve a significant increase in the consequences of a seismic event.

The probability and consequences of a spent fuel cask drop will not be affected by the replacement of the racks. Technical Specification 3.8.C 2 prohibits spent fuel cask movements over any region of the spent fuel pool which contains irradiated fuel. This prohibition will remain effective during and after the storage rack replacement.

Technical Specification 3.8.C.2 prohibits the movements of loads in excess of 2,000 pounds over any region of the spent fuel pit. As the existing and replacement racks weigh in excess of 2,000 pounds, an exception from this prohibition is required for the necessary movements of the racks during the replacement effort. As there is a minimum safety factor of 11.8, for the lifting of a rack by the fuel storage building cask crane, there is a low probability of a drop of a rack. While this temporary exception to Technical Specification 3.8.C.2 does introduce an extremely small possibility of a heavy load drop, which had been previously nonexistent, the consequences of such a drop are within the evaluation criteria of NUREG-0612. Administrative controls will preclude the movement of a rack directly over irradiated fuel. Furthermore, per NUREG-0612, Table 2.1-1 as all of the fuel assemblies stored in the spent fuel pool will have been subcritical in excess of 120 days, the offsite doses will be less than 25% of the 10 CFR Part 100 limits even if the maximum 1345 fuel assemblies are assumed to be damaged as a result of a load drop.

Fuel Enrichment Increase Evaluation

The increased fuel enrichment of up to 4.5 w/o U-235 will not affect the core operating parameters, such as power level, reactor coolant temperature, reactor coolant pressure and core peaking factors. These parameters are considered in detail in the core reload safety evaluations. As such, the operating transient analyses are not impacted solely by a change in the maximum allowable fuel enrichment.

The slightly higher enrichments will facilitate extended fuel cycles. An extended fuel cycle will not increase the fuel rod gap activity since the activity reaches an equilibrium value prior to the end of the current fuel cycle. As such, the off-site dose consequences of a fuel handling accident will not be increased due to an extended fuel cycle.

2. Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response

   The proposed storage rack replacement has been evaluated in accordance with the guidance of the NRC position paper entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Application," appropriate Regulatory Guides and Applicable Standard Review Plan Sections. Additionally, NRC Safety Evaluation Reports on other utilities' applications for storage rack replacement have been reviewed. Based on these evaluations and reviews, the Authority has determined that the installation and use of the replacement racks does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response

   The NRC Staff Safety Evaluation review process has established that the issue of margin of safety when applied to a reracking modification needs to address the following areas:

   1. Nuclear criticality considerations

   2. Thermal-Hydraulic considerations

   3. Mechanical, material and structural considerations

   The established acceptance criterion for criticality is that the neutron multiplication factor in spent fuel pools shall be less than 0.95, including all uncertainties, under all conditions. As discussed in Section II and Enclosure 1, this margin of safety is satisfied by the new rack design.

The methods to be used in the criticality anlaysis conform with the applicable codes, standards and specifications. In meeting the acceptance criteria for criticality in the spent fuel pool, such that K_{eff} is always less than 0.95, including uncertainties at a 95/95 probability confidence level, the proposed reracking of the spent fuel pool does not involve a significant reduction in the margin of safety for nuclear criticality.

Conservative methods are used to calculate the maximum fuel temperature and the increase in the temperature of the water in the spent fuel pool. The thermal-hydraulic evaluation uses the methods described in Section II and Enclosure 1 to demonstrate that the fuel temperature margin of safety is maintained. The proposed reracking results in an increase of the heat load to the spent fuel pool cooling system. The evaluation in Enclosure 1 shows that the existing spent fuel cooling system maintains the pool temperature margins of safety for the calculated increase in heat load. Thus, there is no significant reduction in the margin of safety for thermal-hydraulic or spent fuel cooling concern.

The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all normal and abnormal loadings, such as an earthquake, impact due to a spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object. The mechanical, material, and structural considerations of the proposed racks are described in Section II and Enclosure 1. As also described in Enclosure 1, the proposed racks are to be designed in accordance with applicable portions of the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as modified January 18, 1979; Standard Review plan 3.8.4; and the Indian Point 3 FSAR. The rack materials used are compatible with the spent fuel pool and the spent fuel assemblies. The structural considerations of the new racks address margins of safety against tilting and deflection or movement, such that the new racks do not impact the pool walls, damage spent fuel assemblies, or cause criticality concerns. Thus, the margins of safety are not significantly reduced by the proposed reracking.

Section IV - Impact of Change

This change will not adversely impact the following:

1. ALARA Program
2. Security and Fire Protection Programs
3. Emergency Plan
4. FSAR or SER Conclusions
5. Overall Plant Operations and the Environment

Section V - Conclusion

The incorporation of this change: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

1. U.S. Tool & Die, Inc. Report 8721-00-0033, "Seismic Analysis Report Spent Fuel Storage Racks for Indian Point - Unit 3."

2. NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling."

3. Standard Review Plan 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

4. NRC letter to All Power Reactor Licensees from B. K. Grimes, dated April 14, 1978, entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by NRC letter, dated Janaury 18, 1979.

5. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

6. IP-3 FSAR

7. IP-3 SER