ATTACHMENT I TO IPN-96-121

# PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH THE INCREASE OF FUEL ENRICHMENT TO 5.0 WEIGHT PERCENT URANIUM-235

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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 76 assemblies are to be discharged from the reactor, those assemblies in excess of 76 shall not be discharged earlier than 267 hours 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:

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. The spent fuel storage racks consist of two regions, as shown on Figure 3.8-3: Region 1 (Columns SS-ZZ, Rows 35-64) and Region 2 (Columns A-RR, Rows 1-34). Fuel storage is restricted in each region as follows:

   a. As specified in Figure 3.8-2, fuel assemblies to be stored in Region 2 shall have a minimum burnup exposure as a function of initial enrichment.

   b. As specified in Figure 3.8-1, fuel assemblies to be stored in Region 1 consist of 3 types (Type A, B, C), depending on their initial enrichment and current burnup. Restrictions on location of fuel in Region 1 are as follows:

      1. Type A assemblies may be stored anywhere in Region 1.

      2. A Type B assembly may be stored anywhere in Region 1, provided it is not face-adjacent to a Type C assembly.

      3. Type C assemblies may not be stored in Row 64 or Column ZZ of Region 1. A Type C assembly may be stored in any other Region 1 location provided that all surrounding (face-adjacent) locations are occupied by Type A assemblies, non-fuel components or empty.

D. When any fuel assemblies are in the reactor vessel and the reactor vessel head bolts are less than fully tensioned, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A shutdown margin greater than or equal to 5% $\Delta K/K$

**or**

b. A boron concentration of greater than or equal to 1900 ppm.

The required boron concentration will be verified by chemical analysis daily. With the requirements of the above specification not satisfied, immediately suspend all operations involving core alterations or positive reactivity changes and initiate boration to return to the more restrictive of the limits above.

Basis

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. 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 2400-2600 ppm. 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.

The minimum boron concentration of this water is the more restrictive of either 1900 ppm or else sufficient to maintain the reactor subcritical by at least 5% $\Delta K/K$ in the cold shutdown condition with all rods inserted. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses.

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 with up to 76 assemblies discharged from the reactor.

The waiting time of 267 hours required following plant shutdown before unloading more than 76 assemblies 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 95°F, consistent with the FSAR assumptions(2).

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. 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 filtration 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 ensure 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.

As shown in Figure 3.8-3, the maximum density spent fuel storage racks consist of two regions: Region 1 (Columns SS-ZZ, Rows 35-64) and Region 2 (Columns A-RR, Rows 1-34). Each region has been separately analyzed for close packed storage, where all cells in that region contain fuel of the highest allowable reactivity.

The Region 1 area has also been analyzed for storage of high-enrichment and low-burnup fuel. Figure 3.8-1 categorizes Region 1 fuel assemblies as a function of their initial enrichment and current burnup into Types A, B, and C. Each type has different restrictions as to how it may be stored in Region 1. The least reactive assemblies, which are Type A assemblies, may be stored anywhere in Region 1. The most reactive assemblies, which are Type C assemblies, are stored only in Region 1 with the restrictions of Technical Specification 3.8.C.7.b.3, due to their high reactivity. Type C assemblies cannot be stored face-adjacent to anything more reactive than Type A fuel assemblies. There are no additional restrictions defining storage requirements for diagonally-adjacent fuel assemblies in Region 1. In addition, to prevent a criticality interaction with Region 2 fuel assemblies, Type C assemblies cannot be stored in Column ZZ or Row 64.

The following criteria should be used to categorize Region 1 fuel assemblies. Unburned fuel assemblies at or below 4.2 w/o enrichment are Type A. Unburned fuel assemblies at or below 4.6 w/o enrichment (but greater than 4.2 w/o enrichment) are Type B. Fuel assemblies whose burnup puts them on or above the diagonal line below the Type A zone are defined as Type A.

Fuel assemblies to be stored in Region 2 of the spent fuel 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 satisfies the burnup criterion.

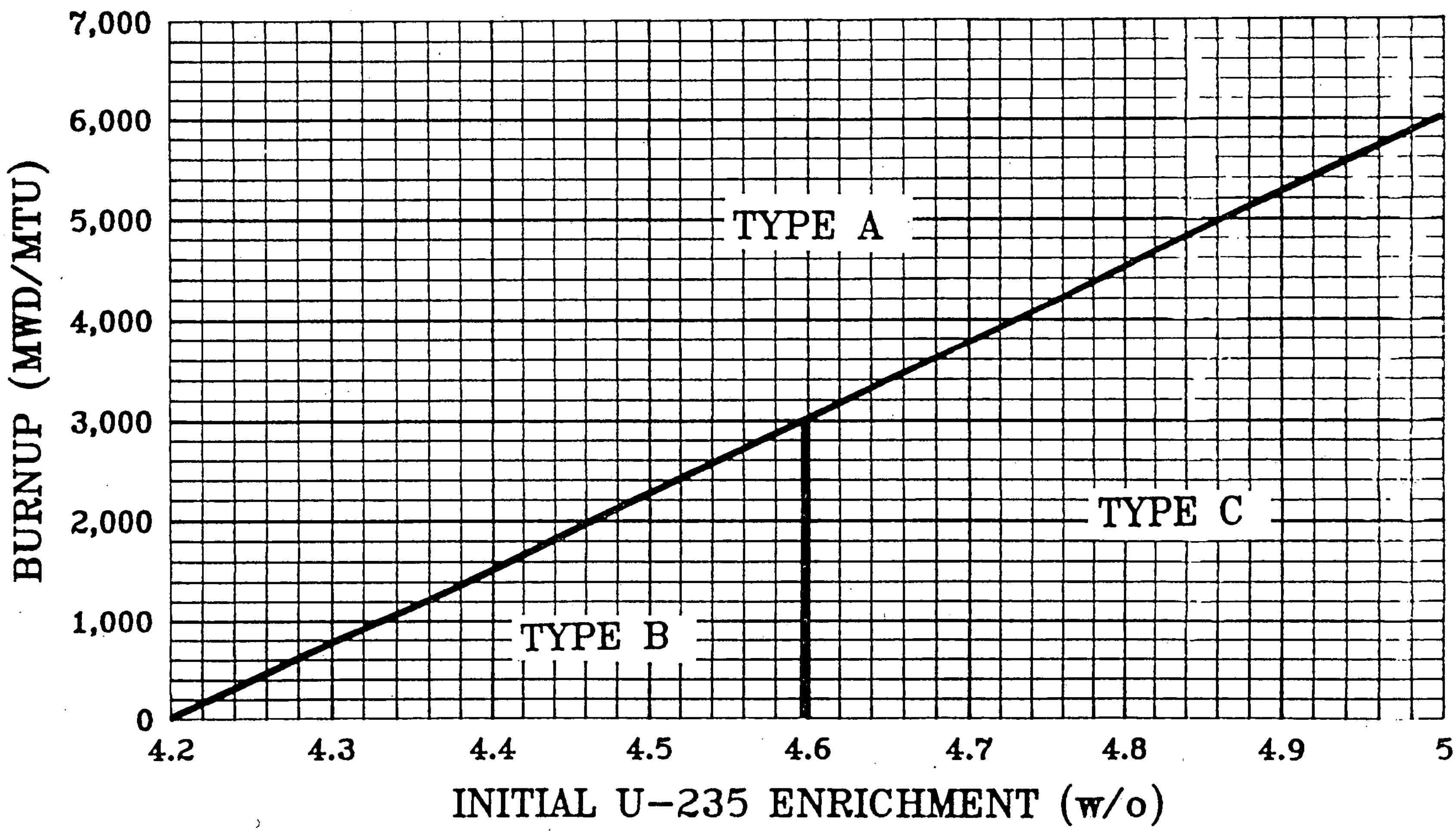
Mechanical stops incorporated on the bridge rails of the fuel storage building crane make it impossible for the bridge of the crane to travel further north than a point directly over the spot in the spent fuel pit that is reserved for the spent fuel cask. Therefore, it will be impossible to carry any object over the spent fuel storage areas north of the spot in the pit that is reserved for the cask with either the 40 or 5-ton hook of the fuel storage building crane. It is possible to use the fuel storage building crane to carry objects over the spent fuel storage areas that are directly east of the spot in the pit that is reserved for the cask. However, the technical specifications and plant procedures prevent any object weighing more than 2,000 pounds from being moved over any region of the spent fuel pit. Therefore, the storage areas directly east of the spot in the pit that is reserved for the cask are protected from heavy load handling by administrative controls.

Dead load tests 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>

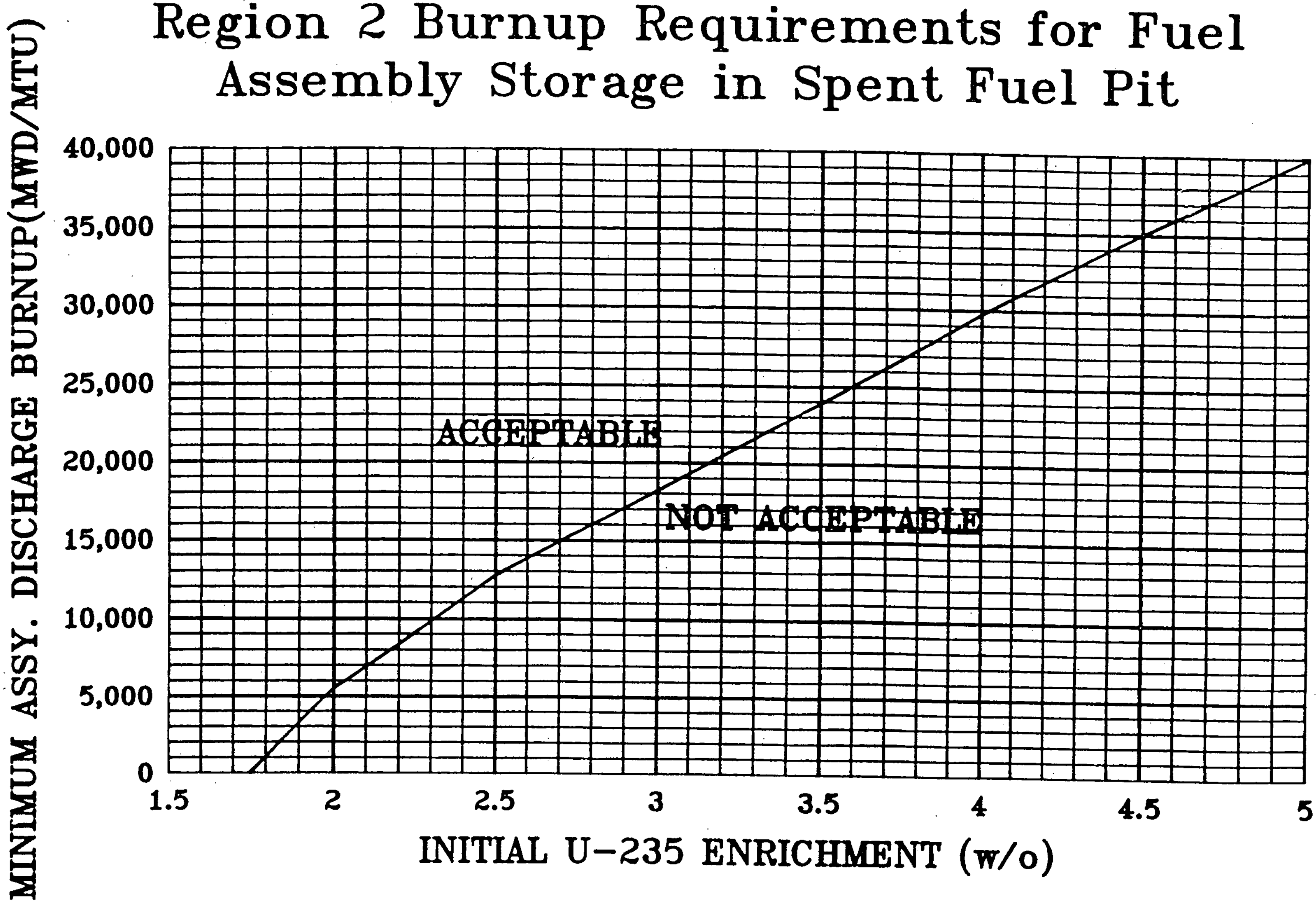
(1) FSAR - Section 9.5.2
(2) FSAR - Section 9.3

# Figure 3.8-1
# Spent Fuel Pit Region 1 Type Definition




**Note: Fresh (unburned) fuel is defined as fuel with a burnup of 0 MWD/MTU.**

# Figure 3.8-2
## Region 2 Burnup Requirements for Fuel Assembly Storage in Spent Fuel Pit



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 system operations.

A. Reactor Core

1. The reactor core contains approximately 89 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO™ tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods,[1] except during Cycle 9 and Cycle 10 operation. For Cycle 9 and Cycle 10 operation only, fuel assemblies W51 and W06 will each contain one stainless steel filler rod in place of a fuel rod.

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 5.0 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 $K_{eff} \leq 0.95$ 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 $K_{eff} \leq 0.95$ under all possible moderation conditions. The capacity of the new fuel racks is 72 assemblies containing fuel pellets enriched to a maximum 5.0 weight percent of U-235 and a maximum $K_{eff}$ (in infinite array) of each fuel assembly of 0.95. Credit may be taken for burnable integral neutron absorbers.

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 pellets enriched to greater than 5.0 weight percent of U-235 shall not be stored in the spent fuel pit or new fuel racks.

ATTACHMENT II TO IPN-96-121

# SAFETY EVALUATION OF THE PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH THE INCREASE OF FUEL ENRICHMENT TO 5.0 WEIGHT PERCENT URANIUM-235

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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DPR-64

SAFETY EVALUATION RELATED TO
PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH THE
INCREASE OF FUEL ENRICHMENT TO 5.0 WEIGHT PERCENT URANIUM-235

## Section I - Description of Changes

This application seeks to amend the Indian Point 3 technical specifications to allow the storage of fuel assemblies with nominal enrichments up to 5.0 weight percent (w/o) uranium-235 (U-235). The major technical specification changes associated with this increased enrichment include:

- revision to fuel pellet enrichment limit from 4.5 to 5.0 w/o U-235; and
- revision to technical specification requirements governing the placement of fuel assemblies in the fuel storage pit.

In addition, this application proposes several administrative changes related to fuel storage to clarify requirements and ensure consistency with other technical specifications and analyses.

## Section II - Evaluation of Changes

Based upon economic and resource considerations, IP3 increased the average fuel cycle length to 24 months. This will result in improved uranium utilization, potential fuel cycle cost savings and a reduced demand on spent fuel storage capacity. In order to achieve a cycle burnup of 24 months, a nominal fuel enrichment of up to 5.0 w/o U-235 is necessary. Such fuel is referred to as extended burnup fuel.

The following discussion provides information in support of the proposed technical specification changes needed to allow the storage of extended burnup fuel. This information has been divided into the following topical areas:

- Summary of NRC Approval of Extended Burnup Fuel Methodology;
- Description of Indian Point 3 Spent Fuel Racks;
- Summary of Indian Point 3 Fresh and Spent Fuel Storage Criticality Analysis;
- Applicability of Spent Fuel Pit Storage Capacity Expansion Analyses, Indian Point 3 License Amendment 90; and
- Applicability of NRC's Generic Environmental Impact Determination.

The proposed increase in fuel enrichment alone will not impact core operating parameters such as power level, reactor coolant temperature, reactor coolant pressure and core peaking factors. These parameters will be evaluated using NRC approved methodology in accordance with Reference 1. As such, they are not discussed in this submittal.

### Summary of NRC Approval of Extended Burnup Fuel Methodology

In response to the commercial nuclear power industry's trend to operate at extended burnups, the NRC requested that each fuel vendor prepare and submit a topical report on their extended burnup experience, methods and test data to provide a generic basis for operation at extended burnups. On July 28, 1982, Westinghouse Electric Corporation submitted WCAP-10125 (Proprietary), "Extended Burnup Evaluation of Westinghouse Fuel." By letter dated October 11, 1985, the NRC issued a Safety Evaluation Report (SER) (Reference 2) on WCAP-10125, which concluded Westinghouse's criteria and analysis methods, as described, were adequate. The NRC SER states:

1.) Fuel damage is not expected to occur as a result of normal operation and anticipated operational occurrences (Condition I and II events);

2.) Fuel damage during postulated accidents (Conditions III and IV events) would not be severe enough to prevent control rod insertion when it is required; and

3.) Core coolability will always be maintained, even after postulated accidents (Condition III and IV events).

WCAP-10125 is applicable to the use of extended burnup fuel at Indian Point 3. As such, the Indian Point 3 reload analysis will be performed utilizing an NRC approved methodology. Consequently, cycle specific parameters will be delineated in future Core Operating Limit Reports (COLRs) per Reference 1 of this attachment, and are not addressed in this submittal.

Description of Indian Point 3 Spent Fuel Racks

The spent fuel storage racks in the Indian Point Unit 3 spent fuel pit are categorized into 2 regions, referred to as Region 1 and Region 2. Currently, Region 1 provides storage for unirradiated fuel with an enrichment up to 4.5 w/o U-235 and provides space for storage of partially burned fuel and a full core offload. Region 2 currently provides storage for irradiated fuel with an initial enrichment up to 4.5 w/o U-235, with restrictions based on assembly burnup.

The spent fuel storage rack design uses a welded honeycomb array of free-standing stainless steel boxes which has no grid frame structure. Each rack is supported and leveled on four screw pedestals which bear directly on the pool floor. Each storage cell in the racks has a welded-in-bottom plate, either 1/2" or 3/4" thick, to support the fuel assembly. A hole in the center of the bottom plate provides for cooling water flow. A narrow rectangular water box surrounds each square storage cell. Boral poison sheets are located between the walls of adjacent storage cells and the water boxes. As such, all storage cells are bounded on four sides by Boral poison sheets, except on the periphery of the pool rack array.

Summary of the Indian Point 3 Fresh and Spent Fuel Storage Criticality Analysis

Enclosure I contains a criticality analysis, prepared by Westinghouse, addressing the storage of 5.0 w/o nominally enriched fresh and burned fuel in the Indian Point 3 fresh and spent fuel storage racks. The analysis assumes fuel assembly parameters based on the Westinghouse 15x15 OFA fuel design, which, under all water density conditions, is the most reactive fuel type in use or storage at Indian Point 3. As such, it bounds all types of fuel currently in use at Indian Point 3.

The analysis used the Indian Point 3 existing fresh and spent fuel storage racks' design and licensing basis criticality limits as acceptance criteria. These limits assume a 95% probability at a 95% confidence level, including all uncertainties. The fresh fuel storage racks' effective neutron multiplication factor, $K_{eff}$, must be less than or equal to 0.95, under optimum moderation conditions (low water density) and at full density flooding conditions. Credit is taken for the effect of burnable integral neutron absorbers (which cannot be separated from the fuel) to maintain $K_{eff}$ at less than or equal to 0.95. The spent fuel storage racks' $K_{eff}$ must be less than or equal to 0.95, including uncertainties, under all conditions.

The analysis identified three fuel handling accident scenarios which could result in the spent fuel pit's $K_{eff}$ exceeding 0.95. They are: 1.) a fuel assembly misload into a position for which the restrictions on location, enrichment, or burnup are not satisfied; 2.) a vertical fuel assembly drop into an already loaded cell; and 3.) placing a fresh 5.0 w/o fuel assembly adjacent to a fully loaded rack. However, per the double contingency principle of ANSI/ANS 8.1-1993, none of these accidents, as defined by the analysis, are credible.

The double contingency principle states it is not necessary to assume two unlikely, independent concurrent events to ensure protection against a criticality accident. However, the subject criticality analysis does assume two such events. Specifically, it first assumes the accident initiating event takes place (i.e., an assembly misload or a dropped assembly). Second, it conservatively assumes the fuel pool water contains no soluble boron. The simultaneous occurrence of both the loss of all soluble boron and an accident initiating event is highly unlikely for two reasons.

First, design and administrative controls exist which make dropping an assembly highly improbable. To preclude dropping an assembly or heavy load over the spent fuel pit, the fuel storage building overhead crane was designed with a minimum safety factor of 5. Dead load tests and visual inspection of the crane prior to fuel handling help to assure it can safely perform its function. In addition, detailed refueling procedures add additional assurance a misload will not take place.

Second, soluble boron concentrations in the spent fuel pit water are maintained sufficiently high as to render an event where no soluble boron is available highly improbable. During fuel handling operations boron concentration is maintained at >1000 ppm. As documented in the subject criticality report, during the most limiting of these scenarios $K_{eff}$ is kept less than or equal to 0.95 with a concentration of 700 ppm of soluble boron in the spent fuel pit with no credit taken for the presence of integral neutron absorbers. If such credit were taken, the boron concentration to maintain $K_{eff}$ less than 0.95 would be considerably less. Although no credit is taken for boron concentration in the criticality analyses, no fuel movement would be initiated unless pit boron concentration is >1000 ppm.

The placement of a fresh 5.0 w/o fuel assembly adjacent to a fully loaded rack is the most limiting of the three scenarios identified as possibly resulting in the spent fuel pit's $K_{eff}$ beyond 0.95. This scenario was analyzed for both Regions 1 and 2. Region 1 does not have sufficient space between the rack and the fuel pool wall to allow accidental insertion. As such, this event is not considered possible for Region 1. However, while highly improbable and beyond the Indian Point 3 design and licensing basis, such an event could take place in Region 2. The analysis showed the limiting impact of such an event on Region 2 is caused when a fresh 5.0 w/o fuel assembly is placed in the open cask area adjacent to 2 sides of the spent fuel racks. Such an event would result in a reactivity increase of 0.11 delta K. Taking credit for a boron concentration of 700 ppm reduces reactivity to below 0.95, as noted above.

The analysis concludes the fresh and spent fuel storage racks' criticality limits can be satisfied for the storage of Westinghouse OFA fuel with a nominal enrichment of 5.0 w/o, if the following configuration restrictions are abided. Assemblies with nominal enrichment of 5.0 w/o may be stored in any location in the fresh fuel rack. Assemblies with nominal enrichments up to 4.6 w/o may be stored in any location in Region 1 of the spent fuel pit without regard to minimum burnup, assuming close packed storage. If checker board storage is assumed for Region I, burned and fresh fuel assemblies may be stored in a 2x2 checkerboard pattern. In this case, the burned cell locations must have an initial enrichment less than 4.2 w/o (nominal) or satisfy the minimum burnup requirements specified in the proposed technical specification Figure 3.8-1. Fuel assemblies stored in the fresh cell locations can have nominal enrichments up to 5.0 w/o, with no requirements for minimum accumulated burnup. Fuel assemblies which may be stored in Region 2, assuming close packed storage, are delineated in the proposed technical specification Figure 3.8-2. There are no restrictions on placement of these assemblies in Region 2.

In an attempt to simplify the wording and implementation of the proposed technical specifications, the Authority has interpreted the enclosed Westinghouse analysis as categorizing Region 1 spent fuel types into three types (Type A, B, and C) as a function of the fuel's initial enrichment and current burnup. Westinghouse has confirmed that this is an acceptable interpretation of the analysis.

For future receipts of new fuel, the Authority will confirm, prior to placing any fuel assemblies in the fresh fuel racks, that the $K_{eff}$ of each new fuel assembly in infinite array is less than or equal to 0.95, with credit taken for the presence of integral neutron absorbers, if necessary. Therefore, the Authority will add limiting fuel assembly $K_{eff}$ to the Reload Safety Analysis Checklist (RSAC).

<u>Applicability of Spent Fuel Pit Storage Capacity Expansion Analyses, Indian Point 3 License Amendment 90</u>

On May 9, 1988, the Authority submitted an application (Reference 3) to revise Appendix A, Sections 3.8, 5.3, and 5.4 of the Indian Point 3 Operating License. The proposed changes would allow the replacement of the high density spent fuel storage racks with maximum density storage racks, increasing the spent fuel storage capability of the spent fuel pit. In addition, the application sought to increase the maximum fuel enrichment allowed in the spent fuel pit and the reactor core from 4.3 w/o to 4.5 w/o U-235. The NRC issued the proposed changes as Amendment 90 to the Indian Point 3 license on October 12, 1989 (Reference 4).

In support of this amendment request, numerous analyses were performed and summarized in Reference 3. Of these analyses, a nuclear criticality analysis addressing the storage of extended burnup fuel and a localized thermal-hydraulics analysis addressing the current service water inlet temperature of 95°F (Reference 12) have been performed. Decay heat loads utilized in the original thermal hydraulic analysis for the maximum density spent fuel pit racks (Reference 5) and for the reevaluation (Reference 12) are consistent with the methodology of NRC Standard

Review Plan 9.2.5, Branch Technical Position ASB 9-2. Therefore, recent NRC concerns on decay heat load calculations (NRC Information Notice 96-39) do not apply.

A detailed spent fuel pit decay heat analysis was prepared to support the spent fuel reracking. A reevaluation of this analysis to address the storage of extended burnup fuel was completed in 1993 and indicated that the decay heat analysis performed in support of the reracking is bounding. In addition, a calculation was recently performed to review the previous analyses and reconfirm that the increased fuel enrichment level of 5.0 w/o U-235 has no effect on the SFP thermal-hydraulics analyses performed to date. A copy of this calculation is included for your review (Enclosure 2). Although the component cooling water (CCW) heat exchangers were replaced in 1992, the effects on spent fuel pit cooling are minimal. Consequently, the calculations supporting the 1988 analysis regarding pressure head effects, general hydraulic profiles, loss of cooling, and time to boil are still valid. As such, the existing spent fuel pit cooling system is adequate for the storage of extended burnup fuel.

The fuel handling analysis, heavy loads evaluation, and seismic event evaluation performed to support the reracking will remain applicable. As these evaluations are based on the structural integrity of the fuel assemblies and racks, as well as that of the crane, the enrichment of the fuel will not adversely affect these evaluations.

The Authority's May 1988 submittal (Reference 3) notes detailed instructions and administrative controls that govern refueling operations, precluding the misload of an assembly or dropped heavy load. Technical Specification 3.8.C.2 prohibits the movement of loads in excess of 2,000 pounds over any region of the spent fuel pit. This prohibition, which is still in place, further ensures that the probability of dropping a heavy load on fuel stored in the spent storage pit remains negligible.

In the NRC's October 12, 1989 SER for Indian Point 3 Amendment 90 (Reference 4), the NRC states the following:

> Since the applicant intends to utilize higher enrichment fuel, for which higher burnups are intended, the staff reanalyzed the fuel handling DBA for this case. Increased burnup could increase offsite doses from the fuel handling DBA by a factor of 1.2 (NUREG/CR-5009, February 1988). Burnup to 60,000 MWD/T would require the use of fuel initially enriched to about 5.3 weight percent U-235. Thus, we conservatively increased the previously estimated doses by a factor of 1.2. In Table 1.0, the new and old DBA doses are presented and compared to the guideline doses in 10 CFR Part 100. As shown in this table, the DBA doses are still well within the regulatory guideline values and are, therefore, acceptable.

As such, the storage of fuel initially enriched to 5.0 weight percent, is bounded by the NRC analysis performed to support Amendment 90.

Applicability of NRC's Generic Environmental Impact Determination

Due to the large number of applications for license amendments permitting incremental increases in the burnup of fuel, the NRC performed an environmental assessment of the use of extended burnup fuel. The NRC's evaluation relied upon the results of a study conducted by Pacific Northwest Laboratories (PNL), the results of which are documented in Reference 6. The PNL study concluded there are no significant adverse effects generated by increasing the present batch-average burnup level to 50 GWD/MTU or above, as long as the maximum rod average burnup of any fuel rod is no greater than 60 GWD/MTU.

On February 29, 1988, the NRC published its determination (Reference 7) than an environmental impact statement need not be prepared for the use of extended burnup fuel in commercial Light Water Reactors (LWRs). The NRC's determination states the use of extended burnup fuel does not pose any significant adverse radiological or non-radiological impacts, nor does it significantly affect the quality of the human environment.

Subsequently, the Office of Nuclear Reactor Regulation (NRR) performed an assessment of the environmental effects of transportation resulting from extended fuel enrichment and irradiation. This assessment is documented as an attachment to Reference 8 regarding amendment requests for extension of fuel irradiation. NRR concluded that no incremental radiological and/or non-radiological impacts would result from the transportation of materials to and from LWR sites.

NUREG/CR-5009 (Reference 6) documents that the amount of long lived radionuclides in the gap of extended burnup fuel is greater than that of less exposed fuel. Correspondingly, the amount of radioactivity that may be potentially released during an accident is increased. However, the projected offsite dose incurred during accidents with extended burnup fuel are still within 10 CFR 100 criteria. Reference 6 concludes that since there is an order of magnitude uncertainty in the risk estimates for accidents, any increased risk from the increased fission products in extended burnup fuel is small.

In addition, a major consideration in each of these studies is the fact that using extended burnup fuel increases the length of the operating cycle, thereby reducing the number of refueling outages. Consequently, there is a corresponding reduction in the estimated number of individuals who may be exposed during the fuel cycle. This reduction in personnel exposure helps to offset the risk associated with the increased quantity of long lived fission products generated in extended burnup fuel.

The Authority has reviewed References 6 and 7 and NRR's environmental assessment of transportation resulting from extended fuel enrichment and irradiation (Reference 11). The Authority concludes that these assessments are applicable to the fuel assemblies being proposed for use at Indian Point 3. As such, the conditions associated with the use of extended burnup fuel at Indian Point 3 are bounded by these NRC studies.

<u>Administrative Changes</u>

This application proposes to make several administrative changes related to fuel storage to clarify requirements and ensure consistency with other technical specifications and analyses.

1. This technical specification amendment updates an overly conservative specification concerning the number of fuel assemblies analyzed for discharge from the reactor. Indian Point 3 is analyzed for a 76 assembly discharge

(References 5 and 12) and the revision to the technical specifications from 72 to 76 assemblies makes them consistent with existing analyses.

2. The specifications are revised to eliminate the term "region" since there is no standard number that defines the amount of new fuel assemblies added during refueling.

3. The designation for "rows" and "columns" used to identify locations in the spent fuel pit have been reversed throughout the technical specifications to agree with technical specification Figure 3.8-3. This reversal of "rows" and "columns" provides for consistency throughout the specifications.

4. The proposed amendment updates the weight of the uranium found in the reactor core. Changing approximately 87 metric tons to 89 metric tons in Section 5.3.A.1 makes this section consistent with current core designs for the 24 month fuel cycle.

5. The proposed amendment updates Basis section 3.8 to account for the increased Hudson river temperature of 95°F, approved by the NRC in Reference 9, and to account for the increased boron concentration in the refueling water storage tank, approved by the NRC in Reference 10.

## Section III - No Significant Hazards Evaluation

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of any accident previously evaluated?

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated. This statement is based on an evaluation of relevant hypothetical accident scenarios, the NRC's evaluation of Westinghouse extended burnup fuel, and the criticality analysis of the Indian Point 3 fresh and spent fuel pits.

Evaluation of Relevant Hypothetical Accident Scenarios

Increasing the enrichment of fuel stored in the spent fuel pit will not increase the probability of occurrence of the following hypothetical accident scenarios:

1. misload of a fuel assembly;
2. spent fuel assembly drop in the spent fuel pit;
3. spent fuel cask drop;
4. loss of spent fuel pit cooling system flow; or
5. seismic event.

1. Misload of a fuel assembly

Detailed instructions and administrative controls govern refueling operations, precluding the misload of an assembly. The proposed storage of extended burnup fuel will not result in these administrative controls being relaxed in any manner. The probability of inserting an assembly into the wrong location is not impacted by the enrichment and burnup of the fuel. Consequently, the proposed changes will not increase the probability of misloading a fuel assembly.

2. Spent fuel assembly drop in the spent fuel pit

The probability of a spent fuel assembly drop in the spent fuel pit is a function of the structural integrity of the fuel storage building overhead crane and the integrity of the crane-assembly coupling. The probability of such a drop is not affected by the enrichment or burnup of the fuel. Therefore, the use and storage of extended burnup fuel will not increase the probability of a fuel assembly drop.

3. Spent fuel cask drop

The probability of a spent fuel cask drop will not be affected by the increased enrichment of the fuel. The probability of such an event occurring is a function of the overhead crane's integrity, which will not be affected by this amendment. In addition, administrative controls are in place to preclude the occurrence of such an event.

4. Loss of spent fuel pit cooling system flow

A reevaluation of the Indian Point Unit 3 decay heat removal analysis to address the storage of extended burnup fuel concluded that the existing spent fuel pit cooling system is adequate to handle the heat load associated with extended burnup fuel since any incremental increase in decay heat for extended burnup fuel is more than compensated for by the greater time interval between refueling outages. In the unlikely event the cooling system should experience a failure, adequate time is available to provide an alternate cooling system, which is not affected by the fuel's enrichment. In addition, an existing off normal operating procedure (ONOP) is available to compensate for any postulated loss of spent fuel pit cooling. Consequently, the storage of extended burnup fuel in the spent fuel pit will not involve a significant increase in the probability or consequences of a loss of cooling system flow event.

5. Seismic event

The enrichment of the fuel has no effect on the probability of a seismic event occurring. In support of Amendment 90 to Indian Point 3's Operating License, a seismic analysis of the spent fuel storage racks was performed. This analysis, which was summarized in Reference 3, is still applicable.

NRC Evaluation of Westinghouse Extended Burnup Fuel

Westinghouse's analysis of the use of extended burnup fuel is documented in WCAP-10125 (Proprietary), "Extended Burnup Evaluation of Westinghouse Fuel". On October 11, 1985, the NRC issued a Safety Evaluation Report (SER) on this WCAP (Reference 2), which concluded that: 1) fuel damage is not expected to occur as a result of normal operation and anticipated operational occurrences (Condition I and II events); 2) fuel damage during postulated accidents (Condition III and IV events) would not be severe enough to prevent control rod insertion when it is required; and 3) core coolability will always be maintained, even after postulated accidents (Condition III and IV events). These conclusions support the determination that the use of extended burnup fuel will not increase the probability or consequences of any accident previously evaluated.

The consequences from accidents involving extended burnup fuel, both during operations and fuel handling, are evaluated in Reference 6. This report, which was the basis for the NRC's determination of no environmental impact, documents the amount of radioactivity released from extended burnup fuel during an accident may be greater than that released from lower burnup fuel. However, the projected offsite dose incurred during accidents with extended burnup fuel is still within 10 CFR 100 criteria. Reference 6 concludes that since there is an order of magnitude uncertainty in the risk estimates for accidents, any increased risk from the increased fission products in extended burnup fuel is small compared to the uncertainties associated with risk estimates. Consequently, the proposed changes do not significantly increase the consequences of any accident previously evaluated.

Criticality Analysis of the Indian Point 3 Fresh and Spent Fuel Pits

Westinghouse performed a criticality analysis of the Indian Point 3 fresh and spent fuel storage racks to determine whether the storage of Westinghouse 15x15 fuel assembly designs with nominal enrichments up to 5.0 w/o U-235 would result in the effective neutron multiplication factor, $K_{eff}$, exceeding design and licensing basis criticality limits. The analysis demonstrated that these criteria would be met during design basis conditions using the fuel storage configurations proposed in this submittal.

Although the analysis identified three scenarios which would exceed the criticality limits, each of these scenarios are outside the design and licensing basis, since they entail the occurrence of two, independent, concurrent events. Specifically, the analysis assumes the occurrence of the initiating accident event and the loss of all soluble boron in the spent fuel pit water. However, the analysis also documents that 700 ppm of soluble boron in the spent fuel pit water will maintain $K_{eff}$ within acceptable limits. The Indian Point Unit 3 spent fuel pit boron concentration is maintained at a minimum of 1000 ppm during fuel handling operations, which is more than adequate to offset the potential reactivity increases incurred from even the most limiting criticality accident scenarios. If credit for integral burnable neutron absorbers is taken, the boron concentration to maintain $K_{eff}$ less than or equal to 0.95 is considerably reduced. Consequently, as supported by the NRC's issuance of similar license amendments to other plants whose criticality analyses have identified similar issues, the proposed amendment does not significantly increase the probability or consequences of any accident previously evaluated.

The administrative changes proposed by this amendment request do not involve a significant increase in the probability or consequences of any accident previously evaluated as they do not involve any plant hardware changes, nor do they change the way the plant systems function.

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

Response:

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. This determination is based on the NRC's SER regarding Westinghouse extended burnup fuel, Indian Point 3 decay heat removal analysis, and spent fuel pit criticality analysis.

The only aspect of the plant that will be physically changed by the proposed amendment will be the enrichment and burnup of the fuel, which will not introduce any new fuel failure mechanisms. While some characteristics of fuel performance change with extended burnup, these considerations have been factored into the design of the fuel. The NRC issued a Safety Evaluation Report (SER) regarding the Westinghouse extended burnup fuel design on October 11, 1985 (Reference 2). In addition, Reference 6 documents that each fuel vendor has adequately considered the performance of extended burnup fuel to preclude the introduction of a new or different type of fuel failure mechanism.

Two site specific evaluations demonstrate the storage of spent and/or fresh extended burnup fuel will not introduce any new fuel storage accidents at Indian Point Unit 3. First, the Authority has verified the existing spent fuel pit cooling system can adequately handle the heat load associated with extended burnup fuel. Second, the criticality analysis performed by Westinghouse demonstrates the criticality limits will continue to be satisfied during design basis conditions. While three scenarios outside of the design basis have been identified as potentially resulting in an increase in spent fuel pit criticality, spent fuel pit soluble boron concentrations are maintained sufficiently high to preclude even the most limiting criticality scenarios from occurring. Consequently, the proposed amendment will not create a new or different kind of accident from any previously evaluated.

The administrative changes proposed by this amendment request do not create the possibility of a new or different kind of accident from any previously evaluated as the changes do not affect current plant configuration or how the plant operates.

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

Response:

The proposed changes do not involve a significant reduction in a margin of safety. This determination is based on the fact that the spent fuel pit racks are not being physically altered, the results of the Indian Point 3 spent fuel pit criticality analysis, the spent fuel pit decay heat analysis, and the NRC issuance of similar amendments to other licensees.

The main safety function of the fresh and spent fuel racks is to maintain the fuel assemblies in a safe configuration through all normal and abnormal conditions. The proposed changes will not result in any changes to the fresh and spent fuel racks or the manner in which they perform. Thus, the margin of safety associated with the fresh and spent fuel racks' ability to physically maintain the fuel in a safe configuration is not significantly reduced by the proposed changes.

A criticality analysis was performed regarding the Indian Point 3 fresh and spent fuel storage racks' ability to store extended burnup fuel within design and licensing basis criticality limits. The analysis concludes during design basis conditions these limits would not be violated. However, it identified three events outside the design and licensing basis which would violate these limits. Nevertheless, if credit is taken for the soluble boron in the spent fuel pit water, criticality is adequately controlled even during these three events. Consequently, as supported by the NRC issuance of similar license amendments to other plants whose criticality analyses have identified similar issues, the proposed amendment does not involve a significant reduction in the margin of safety associated with the control of criticality.

An evaluation was performed to address the spent fuel pit heat load associated with the storage of extended burnup fuel. The analysis concluded the existing spent fuel cooling system will adequately dissipate the heat. Thus, there is no significant reduction in the margin of safety with regards to spent fuel cooling.

The administrative changes proposed by this amendment request do not involve a significant reduction in a margin of safety.

**Section IV - Impact of Changes**

These changes will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR and SER Conclusions
Overall Plant Operations and the Environment

**Section V - Conclusions**

The incorporation of these changes: 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 significantly 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.) NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

2.) NRC letter, Cecil O. Thomas (NRC-Standardization and Special Projects Branch) to E.P. Rahe (Westinghouse - Nuclear Safety Department), regarding acceptance for referencing of Licensing Topical Report WCAP-10125(P), dated October 11, 1985.

3.) NYPA letter, R.E. Beedle to the NRC Document Control Desk, regarding Proposed Changes to Technical Specifications for Spent Fuel Pit Storage Capacity Expansion, dated May 9, 1988 (IPN-88-018).

4.) NRC letter, J. D. Neighbors to J. C. Brons, regarding the issuance of Amendment 90 to the IP3 technical specifications, dated October 12, 1989.

5.) U.S. Tool and Die, Inc., "Thermal-Hydraulic Report - Spent Fuel Storage Racks, Indian Point Unit No. 3," Report No. 8721-00-0104, dated November 24, 1987.

6.) NUREG/CR-5009, PNL-6258, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors", Pacific Northwest Laboratory, February 1988.

7.) Federal Register, Volume 53, Number 39, dated February 29, 1988, pages 6040-6043.

8.) NRC Memorandum, Frank J. Miraglia, Jr. (NRR) to Joseph Scinto (OGC), regarding amendment requests for extension of fuel irradiation and enrichment limits specified in Table 5-4 of 10 CFR 51.52, dated July 7, 1988.

9.) NRC letter, J. D. Neighbors, to J. C. Brons, regarding the issuance of Amendment 98 to the IP3 technical specifications, dated May 7, 1990.

10.) NRC letter, Nicola F. Conicella to Ralph E. Beedle, regarding issuance of Amendment 119 to the IP3 technical specifications, dated June 2, 1992.

11.) NUREG/CR-2325, "Transportation of Radioactive Material (RAM) to and from U.S. Nuclear Power Plants (Draft Environmental Assessment)", Sandia National Labs., Albuquerque, NM., Dec. 1983.

12.) WCAP-12313, "Safety Evaluation for Ultimate Heat Sink Temperature Increase to 95°F," dated July 1989.

13.) IP3 FSAR, Section 9.5

ATTACHMENT III TO IPN-96-121

# AUTHORITY COMMITMENTS FOR THE PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH THE INCREASE OF FUEL ENRICHMENT TO 5.0 WEIGHT PERCENT URANIUM-235

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

**COMMITMENTS ASSOCIATED WITH IPN-96-121**

| Comm. No. | Commitment Description | Due Date |
|---|---|---|
| IPN-96-121-01 | Revise procedures to reflect new enrichment limit and requirements for spent fuel pit storage. (Supersedes IPN-96-092-01.) | 3/13/97 |
| IPN-96-121-02 | Revise FSAR. (Supersedes IPN-96-092-02) | Next applicable FSAR update |
| IPN-96-121-03 | Add to Reload Safety Analysis Checklist (RSAC) the requirements to confirm $K_{eff}$ of all fresh fuel assemblies to be less $\leq 0.95$ including the effects of integral neutron absorbers. | 1/31/97 |

Enclosure 1 to IPN-96-121

# Westinghouse Commercial Nuclear Fuel Division
# Criticality Analysis of the Indian Point Unit 3 Fresh and Spent Fuel Racks

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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