ATTACHMENT I To IPN-86-28
PROPOSED TECHNICAL SPECIFICATIONS

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

6. The fuel storage building emergency ventialtion 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. 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.

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The shutdown margin indicated will keep the core subcritical, even if all control rods were withdrawn from the core.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 120-hour decay time following the subcritical condition and the 23 feet of water above the top of the reactor pressure vessel flange is consistent with the assumptions used in the dose calculation for the fuel-handling accident.

The waiting time of 400 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 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 system is actuated upon receipt of a signal from the area high activity alarm or by a manually-operated switch. The system is tested prior to fuel handling and is in a standby basis.

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.

Fuel assemblies whose initial enrichment is greater than 3.5 w/o U-235 but less than or equal to 4.3 w/o can be stored in the spent fuel pool 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 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-Category 2 fuel. For the purpose of storing Category 2 fuel, non-fuel material or empty locations can be utilized in place of Category 1 fuel.

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.

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

References

(1) FSAR - Section 9.5.2

(2) FSAR - Table 3.2.1-1

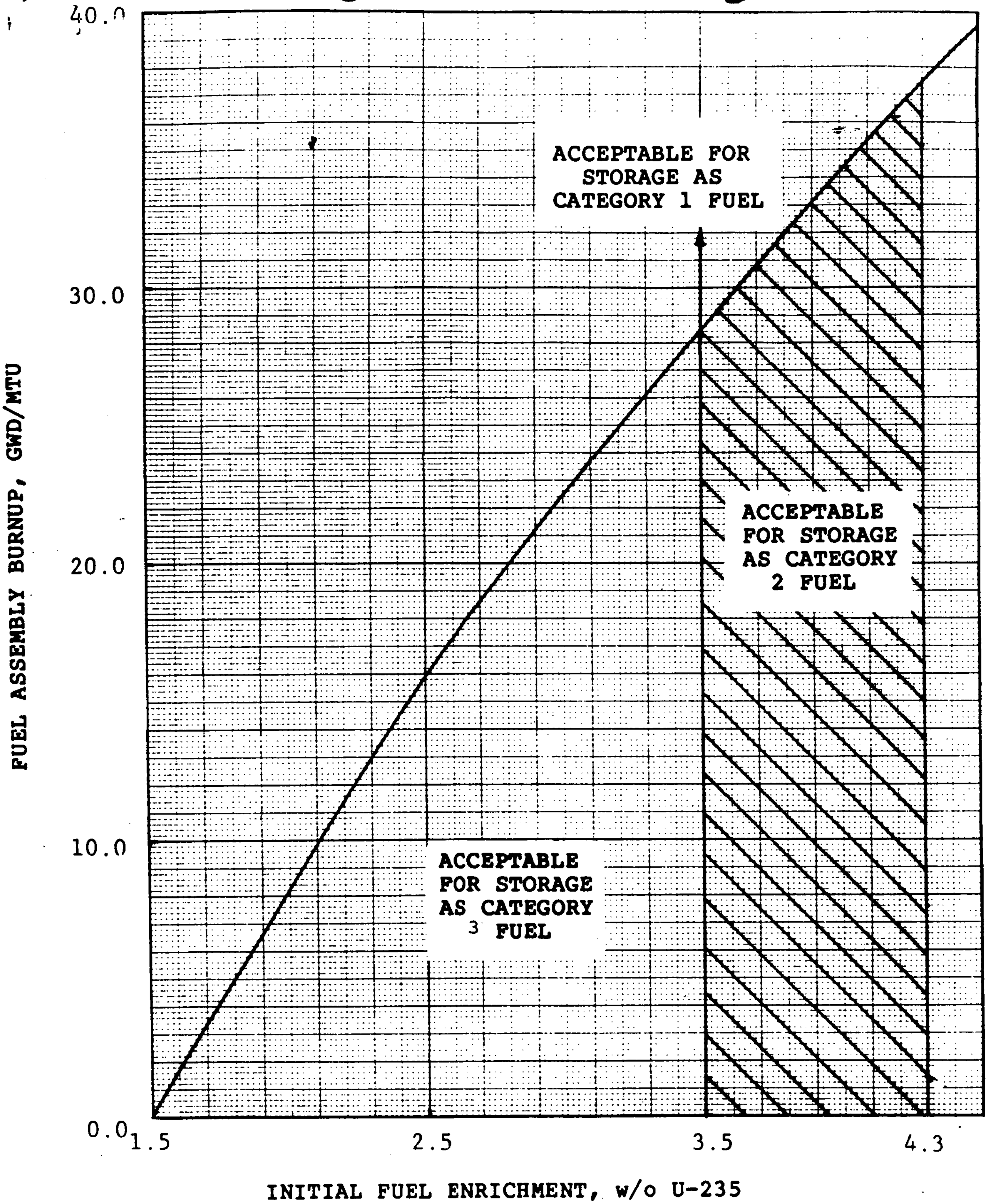


FIGURE 3.8-1

LIMITING FUEL BURNUP VERSUS INITIAL ENRICHMENT

CATEGORY 1 - AREA ALONG THE CURVE AND ABOVE
CATEGORY 2 - AREA BELOW THE CURVE AND $3.5 < \varepsilon \leq 4.3$ w/o U-235
CATEGORY 3 - AREA BELOW THE CURVE AND $\varepsilon \leq 3.5$ w/o U-235

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 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 4.3 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 840 assemblies. The new fuel storage racks are designed to assure $K_{eff} \leq 0.95$ 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 54.6 grams of uranium -235, or equivalvent, per axial centimeter of fuel assembly shall not be stored in the spent fuel pit.

ATTACHMENT II to IPN-86-28
SAFETY EVALUATION

Enclosure 1: Criticality Analysis of Spent Fuel Pool

Enclosure 2: Criticality Analysis of New Fuel Storage Racks

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

Attachment II to IPN-86-28
Safety Evaluation

I. DESCRIPTION OF CHANGE

This revision to the Indian Point 3 Technical Specifications seeks to increase the maximum fuel enrichment to 4.3 w/o U-235 from the current Technical Specification maximum allowable enrichment of 3.4 w/o U-235. The analysis in support of the current Technical Specification maximum allowable enrichment of 3.4 w/o is valid for enrichments up to 3.5 w/o U-235.

II. EVALUATION OF CHANGE

The increased fuel enrichments of 4.3 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 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.

Enclosure 1 provides a detailed analysis of the spent fuel storage racks with fuel enrichments up to 4.3 w/o U-235. The spent fuel can be assigned to one of two categories based on burnup and initial enrichment as specified in Technical Specification 3.8-1. Category 2 fuel will 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. Non-Category 2 fuel can be stored in any location in the spent fuel pool. If the fuel assemblies are loaded in this manner, the analysis has shown that the criticality design criterion of $k_{eff} \leq .95$ will be satisfied.

Enclosure 2 provides a summary of a criticality analysis of the Indian Point 2 new fuel assembly storage racks with fuel stored with enrichments of 4.3 w/o U-235. This analysis applies to the Indian Point 3 new fuel assembly storage racks since the racks are of the same design for both units.

It should be noted that the basis for Technical Specification 3.8.C.2, which appears on page 3.8-5, has been revised to accurately reflect that Technical Specification.

III. NO SIGNIFICANT HAZARDS EVALUATION

In accordance with the requirements of 10 CFR 50.92, the application has been determined to involve no significant hazards based upon the following:

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

Response

The increased fuel enrichment of up to 4.3 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 detailed 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 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.

In conclusion, the proposed Technical Specifications change for maximum allowable enrichment and fuel storage will not increase the probability or consequences of the FSAR design basis accidents.

(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 change seeks to increase the enrichment of the fuel pellets only. No hardware changes are necessary. The maximum power operation level will not be increased. As such, the requested change will not create a new or different kind of accident.

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

Response

The analysis provided by Enclosure 1 shows that the criticality design criteria of $k_{eff} \leq .95$ will not be exceeded if the fuel is loaded into the spent fuel cells per Technical Specification 3.8.

IV. Impact of Change

This change will not adversely impact the following:

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

V. Conclusion

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 possiblity 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 basis for any Technical Specification; d) does not constitute an unreviewed safety question; and e) invloves no significant hazards considerations as defined in 10 CFR 50.92.

VI. References

a) IP-3 FSAR
b) IP-3 SER

ENCLOSURE 1

CRITICALITY ANALYSIS OF SPENT FUEL POOL

NEW YORK POWER AUTHORITY

INDIAN POINT 3 NUCLEAR POWER PLANT

DOCKET NO. 50-286

TABLE OF CONTENTS

1. SUMMARY

A detailed nuclear analysis has been completed for the existing spent fuel storage racks at the Indian Point Generating Station Unit No. 3 (IP3). The objective of this analysis was to demonstrate that the existing spent fuel storage racks can safely store unirradiated 15X15 Westinghouse fuel assemblies with initial enrichment of up to 4.3 w/o U-235, provided the fuel is loaded as specified in Section 4.0 of this report. It should be noted that this analysis is for the spent fuel racks which are currently in place and that no hardware or material changes are proposed.

The analysis was performed assuming selective loading of unirradiated fuel with enrichments up to 4.3 w/o U-235 in a checkerboard manner. This analysis takes credit for fuel depletion in the calculation of the fuel/rack reactivity state. In order to store unirradiated fuel with enrichments up to 4.3 w/o U-235, certain conditions must be met. These include the requirement that the storage cells adjacent to the four faces of the cell designated for the higher enrichment fuel must contain irradiated fuel which has accumulated a specific burnup, or non-fuel material. The burnup level depends on the initial enrichment of the fuel assemblies. In this manner the reactivity of the assembly array is limited by restricting the allowable reactivity of cell contents directly adjacent to new fuel assemblies with initial enrichments of up to 4.3 w/o U-235.

The analysis contained in this report is intended to supplement the previous criticality analysis which supported the license amendment for the existing spent fuel storage racks[1]. In the previous analysis the principal method of calculation used to determine the k_{eff} of the Indian Point (IP3) spent fuel storage racks was the Monte Carlo Codes KENO-III[2] and KENO-IV[3,4]. In the present analysis the exposure level as a function of initial fuel enrichment for irradiated fuel was determined with an explicit PDQ-7[5] model. Macroscopic cross sections for the PDQ-7 model as a function of enrichment and burnup were developed with CASMO-2E[6]. The CASMO/PDQ method has been benchmarked against KENO calculations.

Accounting for all uncertainties, the k_{eff} for the Type A and Type B fuel storage racks as determined from this analysis are 0.932 and 0.937, respectively. These values meet the criticality design criterion of $k_{eff} \leq .95$ and are substantially below 1.0. It was therefore concluded that both the Type A and Type B spent fuel storage racks at Indian Point No. 3, when loaded with fuel as specified in Section 4.0 of this report, are safe from a criticality standpoint.

2. INTRODUCTION

The nuclear analysis described in this report demonstrates that with fuel loaded as specified in Section 4.0 fuel assemblies with enrichments up to 4.3 w/o U-235 can be stored with the k_{eff} of the system conservatively calculated to be less than 0.95. The analysis is based on conservative assumptions with respect to pool water temperature and conditions, fuel geometry, etc. In addition to the reference configuration, fuel misloading incidents were also analyzed. In this case, it was assumed that the fuel racks were completely loaded with unirradiated fuel with enrichment of 4.3 w/o U-235 and the k_{eff} of the fuel racks with and without soluble boron in the pool water was determined.

The following sections of this report describe the general arrangement of the existing fuel storage racks, methods for criticality analysis, results of the calculations and benchmarking of the methods.

3. GENERAL ARRANGEMENT AND CONFIGURATION OF THE EXISTING SPENT FUEL STORAGE RACKS

The general arrangement of the existing spent fuel storage racks at IP3 have been described previously(1) and a brief description is included herein to aid in interpreting the results of the current analysis. As shown in Figure 3-1, the IP3 spent fuel storage pool contains 12 storage modules of various sizes ranging from 6X6 to 9X9 which contain individual storage cells for 36 to 81 fuel assemblies. The rack modules have been installed such that the center-to-center spacing of fuel assemblies in adjacent modules is 14.0". The total number of fuel storage locations is 840 (including 3 locations for failed fuel storage).

In the upper section of the pool, the region above the dashed line in Figure 3-1, 459 storage locations are provided (excluding the 3 cells for failed fuel). In this region the individual fuel storage cells consist of a square stainless steel tube with nominal inside dimensions of 9.0 inches and a nominal wall thickness of 0.170". Two 0.125" thick borated (1.0 w/o natural boron) stainless steel plates (7.0" wide and 145" long) are affixed on two adjacent sides to the outer surfaces of the stainless steel guide tube as shown in Figure 3-2. The stainless steel guide tubes are arranged in a square pitch with 12.0" nominal center-to-center spacing. These storage modules are designated as Type A racks.

In the lower section of the pool, modules designated as Type B provide storage capacity for 328 fuel assemblies. The individual stainless steel guide tubes in the Type B modules have the same dimensions and material structures as the Type A guide tube, except that an additional borated stainless steel plate is used as shown in Figure 3-2. The Type B storage cells are arranged in a rectangular pitch, 11.25" X 12.0". In both the upper and lower sections, those guide tubes located on the periphery of each module do not have poison plates on the sides facing the 5.0" water gap between modules or the water gap between the module and the pool liner.

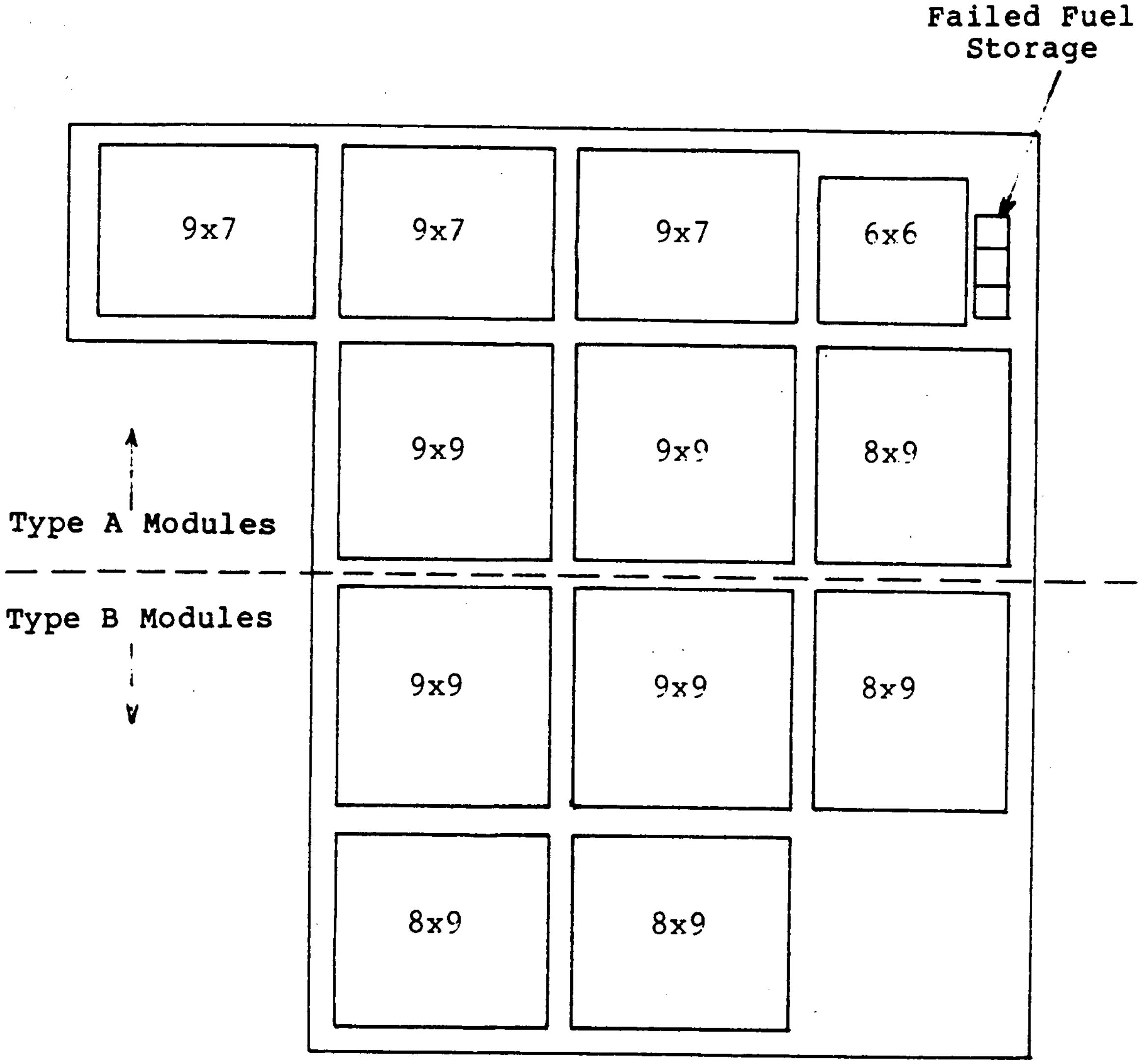
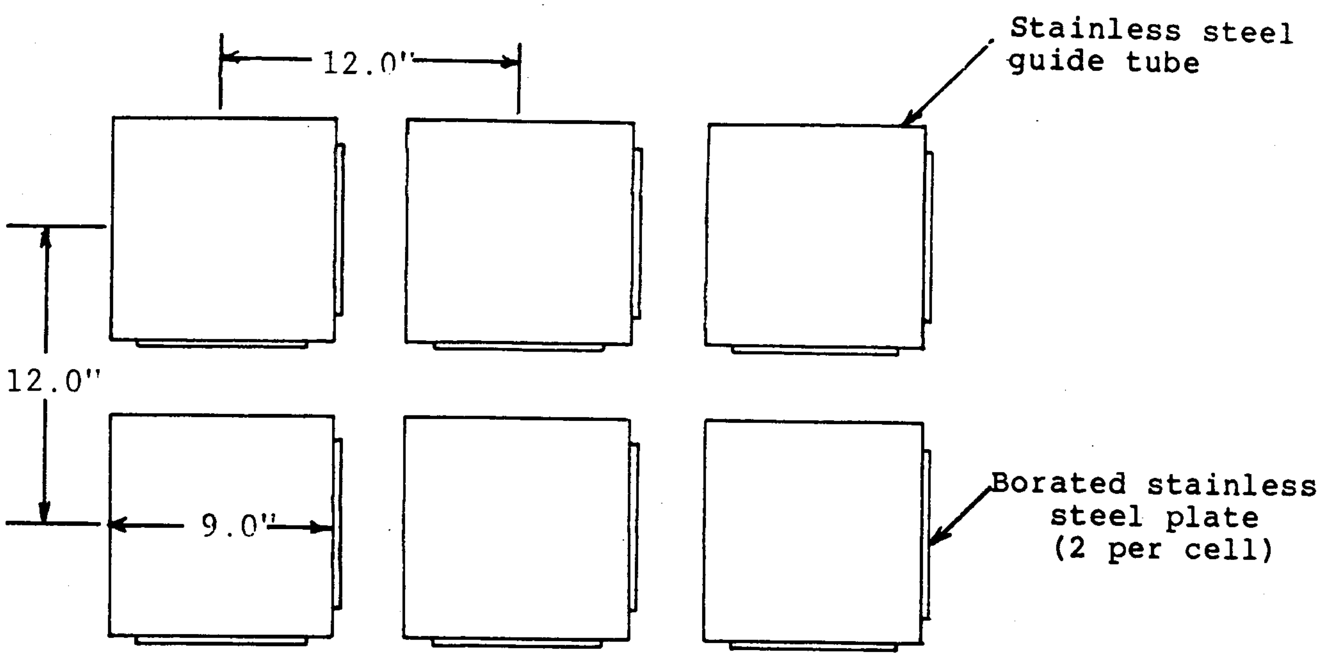


FIGURE 3-1

<u>INDIAN POINT 3 SPENT FUEL STORAGE POOL:</u>
<u>SPENT FUEL STORAGE RACK ARRANGEMENT</u>

Total spent fuel storage capacity	837 locations
Total failed fuel storage capacity	3 locations
Total capacity	840 locations

TYPE A CELLS



TYPE B CELLS

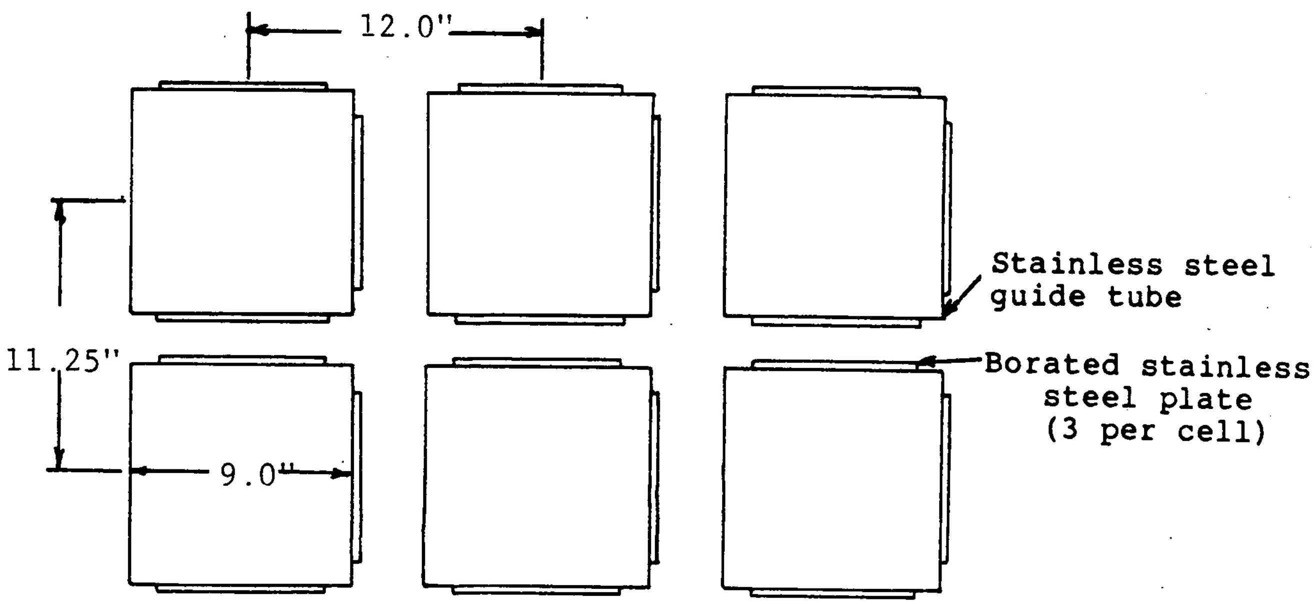


FIGURE 3-2

TYPE A AND TYPE B STORAGE CELLS
(Nominal Dimensions)

4. FUEL STORAGE CONFIGURATION

The nuclear analysis which supports the existing Plant Technical Specifications[1] demonstrates that fuel with initial enrichments of up to 3.5 w/o U-235 and zero exposure can be safely stored in all storage cells of the existing IP3 spent fuel storage racks. That analysis is based on the assumption of a repeating array of 3.5 w/o U-235 assemblies in the x-y directions.

In order to increase the enrichment of fuel which can be stored in the existing spent fuel storage racks, two categories of fuel are defined, category 1 and category 2. Category 1 fuel is defined as having combinations of initial fuel enrichment and exposures above and to the left of the curve in Figure 4-1. Category 2 fuel is defined as having initial fuel enrichment between 3.5 w/o and 4.3 w/o U-235 and corresponding exposures below and to the right of the curve in Figure 4-1 as represented by the cross-hatched portion. In order to store category 2 fuel in the spent fuel storage racks, category 1 fuel must be loaded into the racks in a checkerboard fashion with the intermediate storage cells reserved for category 2 fuel. In this manner, the reactivity of the assembly array is limited by requiring that the reactivity of the cell contents directly adjacent to category 2 assemblies be less than or equal to the reactivity of depleted fuel (category 1). Alternatively, category 2 fuel may be stored with non fuel materials or water only in the four storage locations adjacent to the four of faces the cell reserved for the category 2 fuel. Any noncategory 2 fuel can be stored on a repeating array of assemblies in the x-y directions.

The supplemental nuclear analyses described in this report are the basis for the constant rack reactivity curve shown in Figure 4-1.

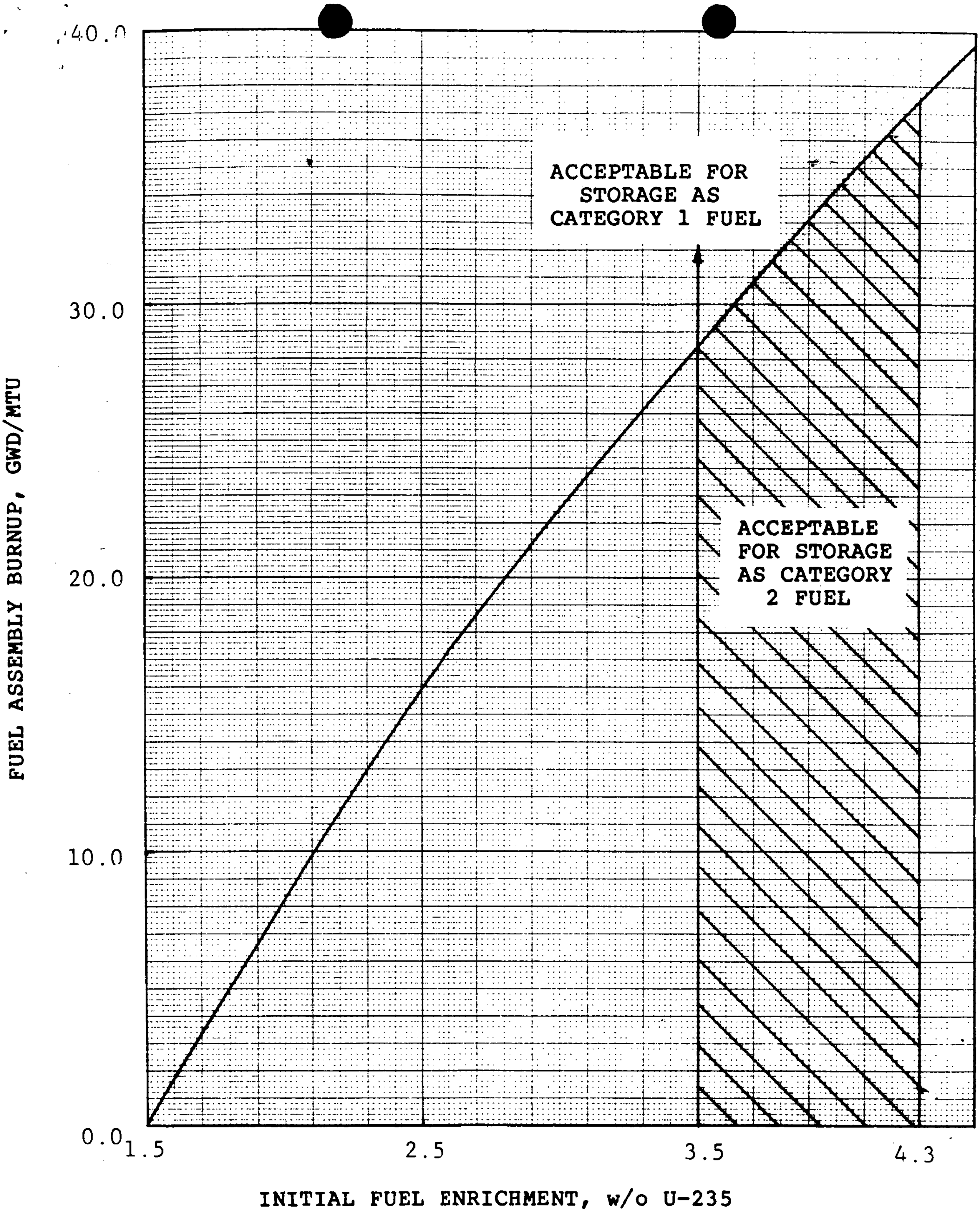


FIGURE 4-1

LIMITING FUEL BURNUP VERSUS INITIAL ENRICHMENT

CATEGORY 1 - AREA ALONG THE CURVE AND ABOVE
CATEGORY 2 - AREA BELOW THE CURVE AND $3.5 < \varepsilon \leq 4.3$ w/o U-235

5. NUCLEAR CRITICALITY ANALYSIS

5.1 NUCLEAR DESIGN BASIS, ASSUMPTIONS AND METHODS

To assure that the k_{eff} of the IP3 spent fuel racks is less than 0.95 when fully loaded with fuel, the maximum enrichment permitted is 4.3 w/o U-235 for unirradiated fuel provided the fuel is loaded as specified in Section 4.0. With all uncertainties included, there is a 95% probability at a 95% confidence level that the effective multiplication factor is less than .95 as recommended in ASNI N210-1976 and the NRC document "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (7).

The analysis which has been performed utilizes the following conservative assumptions in demonstrating that the design basis has been met:

1. The pool water was assumed to have a density corresponding to 68°F.
2. Soluble boron in the pool water is not considered.
3. Neutron absorption in the fuel assembly grid spacers is not considered.
4. No credit is taken for burnable poison fixtures.
5. The analysis assumes that the fuel and rack array are infinite in the axial and lateral directions.
6. The 15 X 15 fuel assembly is assumed to be of the

Westinghouse 15 X 15 Optimized Fuel Assembly (OFA) with Zircaloy guide tubes.

In addition, the present analysis is intended to be an extension of the reference analysis[1]. The present analysis assesses the trade-off of fuel enrichment in one half of the fuel storage locations against fuel depletion and/or enrichment in the other one half of the fuel storage locations. For the purpose of the current analysis it has been assumed that uncertainties attributable to "worst case" stack up of rack dimensional variations and "worst case" moderator temperature effects developed in the reference analysis apply. This assumption is justified since neither the rack structure nor materials have changed.

In addition, in the present analysis it is conservatively assumed that uncertainties in the reactivity of the depletion dependent fission products and other isotopics introduce an additional uncertainty in the rack k_{eff} of 0.02 as discussed in Section 5.4. The methods utilized in the present analysis include the use of CASMO-2E and PDQ-7 to assess the trade-off of initial fuel enrichment <u>versus</u> fuel depletion. CASMO-2E models of the Type A and Type B fuel/racks were utilized to provide macroscopic cross sections for the non-fuel (rack structure, poison and water exterior to the fuel assembly) for use in a PDQ-7 model of the rack and fuel. Depletion dependent macroscopic cross sections as a function of initial fuel enrichment were developed with a 1/8 assembly CASMO model of the IP3 fuel assembly. Using the depletion dependent isotopics from this model a series of restart calculations in the cold condition, with no xenon or soluble boron, were performed to provide fuel macroscopic cross sections as a function of initial fuel enrichment and burnup. These cross sections were subsequently used as input to PDQ models of the Type A and B racks to determine, iteratively, fuel burnup as a function of initial enrichment which provides a constant rack k_{eff}. A detailed description of the models is contained in Section 5.2.

5.2 FUEL RACK MODELS AND METHODS OF ANALYSIS

5.2.1 Reference Fuel Assembly Design Parameters

All models described subsequently are based on the Westinghouse 15 X 15 OFA fuel assembly. This assembly is characterized by a 15 X 15 array of fuel rods with 20 rods replaced with control rod guide tubes and the central rod replaced with an instrumentation thimble. Table 5-1 summarizes the IP3 fuel design parameters.

5.2.2 CASMO Fuel Rack Models

CASMO fuel/rack models--in which all fuel assembly components, rack structure, inner and outer water gaps and poison are represented explicitly--were developed for the Type A and B storage cells as shown in Figures 5-1 and 5-2. It should be noted in these Figures that a limitation in the existing version of the CASMO program is that the borated stainless steel must be assumed to completely cover the face of the square stainless tube of the storage cell. The borated stainless steel on IP3 racks does not completely cover the face of the tube as shown in Figure 3-2.

The purpose of the CASMO rack model is to provide macroscopic cross sections for the water regions, borated stainless steel and stainless steel structure for subsequent use in a PDQ model of the IP3 Type A and B storage cells. In this respect, the H factor option available in CASMO-2E was used to develop transport corrected cross sections in the borated stainless steel rack regions for PDQ. By using the H-factor option, the macroscopic absorption cross sections in the poison plates are adjusted so that the k_{eff} of the PDQ diffusion theory calculation matches the k_{eff} provided by the transport theory (CASMO) calculation. CASMO first completes the transport calculation and then performs a series of diffusion theory calculations (DIXY) which are identical to those performed by PDQ. During each subsequent DIXY calculation, the macroscopic absorption cross sections in the borated stainless steel are adjusted until the k_{eff} of the DIXY calculation matches the transport calculation. In this manner, the diffusion theory bias (generally a 2-3% underprediction of k_{eff} in regions containing black absorbers) is eliminated and the diffusion theory calculation is normalized to the more exact transport theory calculation. The accuracy of this method has been demonstrated via CASMO and PDQ benchmark calculations for many critical experiments[8] as well as for operating cores with control rods[9].

5.2.3 CASMO Fuel Assembly Model

In order to generate macroscopic cross sections as a function of initial fuel enrichment and fuel assembly exposure, a 1/8 assembly CASMO model was used. In this model all fuel rods, guide thimbles and the narrow water gap between assemblies are represented explicitly as shown in Figure 5-3. The model was depleted at hot full power reactor conditions and isotopic concentrations were retained at various burnup steps. The procedure was repeated several times for fuel assembles with initial enrichments over the range of 1.5 to 4.3 w/o U-235. Subsequently, using the isotopic concentrations as a function of exposure so developed, restart calculations were performed at cold zero power conditions with zero xenon and zero soluble boron. These restart calculations provide fuel assembly average macroscopic cross sections for the fuel regions in the PDQ model.

5.2.4 PDQ-7 Fuel/Rack Models

In order to assess the trade-off between fuel enrichment in one half of the assemblies stored in the fuel racks against fuel depletion and/or enrichment in the other half of the assemblies, the PDQ models shown in Figures 5-4 and 5-5 were used. All regions of the rack structure, borated stainless steel and water gaps have been represented explicitly. The spatial mesh distribution in the PDQ models are identical to that in the CASMO rack models, consistent with the use of the H factor option described previously.

The PDQ model was first applied to determine the fuel rack reactivity with one half the assemblies containing fuel of initial enrichment of 4.3 w/o U-235 (zero burnup) and the other half of the assemblies at a lower enrichment of 1.5 w/o U-235 (zero burnup). Subsequently, the 1.5 w/o fuel was replaced with fuel of intermediate enrichments which had achieved some level of exposure. The exposure was varied iteratively until the k_{eff} of the rack with irradiated fuel matched the k_{eff} of the rack with one half the assemblies at 4.3 w/o U-235 at 0 GWD/MTU and the other half of the assemblies at 1.5 w/o U-235 at 0 GWD/MTU. This process was repeated as a function of initial enrichment to develop a curve of fuel assembly exposure versus initial enrichment as shown in Figure 4-1. The curve in Figure 4-1 represents constant rack reactivity with 4.3 w/o U-235 fuel at zero burnup loaded in 1/2 of the rack locations and category 1 fuel in the other half of the locations.

5.3 RESULTS OF THE CRITICALITY ANALYSIS

5.3.1 Constant Rack Reactivity Calculations

Analysis with the PDQ models described previously was completed to determine the intercept (0 burnup) point of the constant reactivity curve shown in Figure 4-1. For this analysis, it was assumed that one half of the storage locations were filled with 1.5 w/o fuel and the other half filled with 4.3 w/o fuel, both fuels assumed to be unirradiated. The best estimate of k_{eff}'s were calculated to be 0.901 and 0.906 for the Type A and Type B racks, respectively.

Using these points as the reference, fuel with higher initial enrichments and which had accumulated some burnup was substitured for the 1.5 w/o U-235 fuel assemblies. The burnup of this fuel was varied until a fuel/rack reactivity of k_{eff} = .901 was obtained for the Type A racks and k_{eff} = .906 for the Type B racks. Using this procedure, a specific exposure for a fuel with a given intial enrichment was determined such that the fuel/rack k_{eff} is constant for each of the fuel rack designs.

Table 5-2 contains the exposure level for fuel assemblies with initial enrichments from 1.5 to 4.3 w/o U-235 which provides constant fuel/rack reactivity. These analyses are the basis for the curve shown in Figure 4-1 which defines the category 1 and 2 fuel types.

5.3.2 Checkerboard Loading of 4.3 w/o U-235 Fuel

The fuel/rack reactivity of the Type A racks was determined with the PDQ model assuming 4.3 w/o U-235 fuel loaded in every other location with the alternate cells vacant (water filled). The effect of water temperature variations from 68^{o}F to 212^{o}F was also determined. The Type A racks were selected for these analyses since they contain less borated stainless steel than the Type B racks. Accordingly, increases in pool water temperature potentially could increase the fuel/rack reactivity relative to the reference condition at 68^{o}F. The results of these analyses shows that the k_{eff} of the fuel/rack configuration is less than .90 and maximum reactivity occurs at a water temperature of 68^{o}F.

5.3.3 Fuel Misloading

As a worst case upper bound analysis, the inadvertent loading of unirradiated 4.3 w/o fuel in every location of the Type A and Type B fuel storage racks was considered. For the initial calculations it was assumed that a zero soluble boron concentration condition prevailed in the pool water. The calculations were repeated with 500 ppm soluble boron in the pool water.

This highly unlikely worst cast conditon was analyzed using the PDQ fuel rack model described previously. With zero ppm soluble boron and a uniform loading of 4.3 w/o U-235 fuel assemblies, the k_{eff}'s are 0.962 and 0.961 for the Type A and B fuel racks, respectively. It is therefore concluded that for this multiple failure condition (i.e., complete loading of the racks with 4.3 w/o fuel as well as no soluble boron in the pool) the fuel/rack configuration is still subcritical by more than 0.03. If the soluble boron concentration is 500 ppm, the k_{eff} of both the Type A and B racks is calculated to be 0.888.

5.4 Benchmarking

The CASMO/PDQ method employed for the analysis of the Indian Point 3 fuel storage racks has been benchmarked against KENO calculations for PWR fuel storage racks. The KENO calculations have been completed by independent organizations and previously submitted and approved by the USNRC.

Table 5-3 summarizes the fuel design parameters and fuel storage rack construction details for each of the benchmark calculations. Five of the cases are for 15x15 fuel types identical to or very similar to that at Indian point Unit 3. The range of enrichments is 1.42 to 4.5 w/o U-235 and bound those considered in the analysis of the Indian Point 3 spent fuel storage racks. The poison loadings, cell spacing and structural characteristics of the racks also bound the IP3 fuel storage racks.

The k_{eff} for each of the fuel rack designs as calculated by the CASMO/PDQ method are compared with the KENO calculations in Table 5-4. Of the KENO calculations in Table 5-5, all have been calculated with KENO-IV using 123 energy group NITAWL-XSDRN cross sections except the Indian Point Unit No. 3 Type A fuel rack and the Crystal River fuel rack. The IP3 Type A rack was analyzed with KENO-III (Reference 2) using 18 energy group cross sections. The Crystal River case was analyzed using the 16-energy groups Hansen-Roach cross section set. Accordingly, these two cases are included for comparison only and have not been used for the determination of CASMO/PDQ model uncertainty.

In general, the CASMO/PDQ method tends to overpredict the KENO calculations with the mean bias being +0.003. The standard deviation for these six sets of calculations is +/- 0.004. Analysis of critical experiments by others[8] with CASMO/PDQ and KENO-IV have shown similar results.

For a sample size of 6, the one-sided tolerance factor is 3.707 for a 95% probability--95% confidence level. The appropriate model uncertainty at this level of probability and confidence level is therefore 3.707x.004 = 0.015.

To verify that the 0.020 uncertainty attributable to burnup dependent isotopics and cross sections is conservative, the CASMO depletion calculations were independently confirmed with the EPRI-CELL code[16]. The EPRI-CELL cross sections have in turn been used in a PDQ core model for the purpose of core follow at the Indian Point Unit 2 reactor[18]. Comparisons have been made over seven cycles of operation between predictions using PDQ with the EPRI-CELL cross sections and actual core operating data such as soluble boron concentration versus core burnup.

These comparisons have verified that the EPRI-CELL methods provide predictions of core depletion characteristics to within less than 1.0% in core reactivity.

Using EPRI-CELL, a 4.3 w/o u-235 fuel rod was depleted using the same power-exposure and soluble boron history as was used in the CASMO analysis. The difference in infinite multiplication factor between CASMO and EPRI-CELL has been calculated and shows that at all burnup steps, CASMO overpredicts the EPRI-CELL value (mean overprediction = +0.006). It is therefore concluded that CASMO provides a conservative prediction of fuel reactivity relative to EPRI-CELL.

To verify the accuracy of EPRI-CELL, operating data over seven fuel cycles have been compared with EPRI-CELL/PDQ calculations(17). Over the seven fuel cycles, mean difference in core reactivity is -0.003 with EPRI-CELL/PDQ underpredicting the core reactivity slightly.

The benchmarking of CASMO versus EPRI-CELL has demonstrated that fuel reactivity as calculated by CASMO is conservative relative to the EPRI-CELL. The comparison of the EPRI-CELL calculations with operating data shows that the EPRI-CELL cross sections provide an accurate prediction of core depletion (a mean bias of -.003 in core reactivity over seven fuel cycles) and thus burnup dependent fuel isotopics and cross sections. It has been demonstrated, therefore, that the value of 0.020 assumed to be the uncertainty attributable to the burnup dependent isotopics and cross sections is conservative.

5.5 MODEL UNCERTAINTIES, BIASES AND CONSERVATISMS

In the original criticality analysis [1,18] for the IP3 spent fuel storage racks, variations in rack k_{eff} resulting from rack manufacturing tolerances and pool water temperature were determined. In that analysis, the combined effects of worst case stack-up of manufacturing tolerances (geometry effects) and worst case pool temperature effects were calculated. These "worst case" geometry conditions include:

Minimum storage cell pitch
Minimum storage cell inside dimension
Minimum storage cell wall thickness
Four adjacent fuel assemblies at positions of closest proximity (eccentric position)

The "worst case" pool temperature condition was determined as follows:

Water inside the fuel assembly at 68^{o}F
Water between assemblies at 212^{o}F

This condition, while not realistic, was determined to maximize the reactivity of the fuel/racks.

The results of the calculated k_{eff} for nominal conditions and for worst case geometry/temperature conditions are shown in Table 5-5. The Type A rack was analyzed with KENO-III while the Type B rack was analyzed with both KENO-III and KENO-IV. Since the KENO-III results for the Type B rack maximimze the variations in k_{eff} due to worst case geometry and temperature, they will be used for the calculation of the 95% probability--95% confidence level k_{eff} of the Type B fuel rack.

The final k_{eff}, including all uncertainties, at a 95% probability and 95% confidence level, is calculated as:

$$k_{eff} = k_{nom} + B_{meth} + (\Delta k^2_{geom} + \Delta k^2_{temp} + \Delta k^2_{meth} + \Delta k^2_{bu})^{1/2}$$

where:

k_{nom} = best estimate k_{eff}

B_{meth} = biases in the CASMO/PDQ method

Δk_{geom} = worst case variations due to geometry effects

Δk_{temp} = worst case variations due to temperature effects

Δk_{meth} = method uncertainty = 0.015

Δk_{bu} = uncertainty attributable to burnup dependent isotopics and cross sections = 0.020

In Section 5.4 it was shown that the method bias is positive (CASMO/PDQ method overpredicts reactivity by + 0.003). For conservatism it will be assumed that the method bias is zero. At 95% probability and 95% confidence level, the k_{eff} of the Indian Point 3 Type A and Type B spent fuel storage racks are calculated to be .932 and .937, respectively, both of which are less than the 0.95 criterion.

TABLE 5-1

FUEL ASSEMBLY DESIGN PARAMETERS

Fuel Rod Data	
Outside dimension, in.	0.422
Cladding thickness, in.	0.0243
Cladding material	Zr-4
Pellet diameter, in.	0.3659
UO_2 density, % T.D.	94.5
UO_2 stack density, g/cm^3	10.357
Maximum enrichment, wt. % U-235	4.3
Fuel Assembly Data	
Number of fuel rods	204
Fuel rod pitch, in.	0.563
Control rod guide tube	
Number	20
O.D., in.	0.533
Thickness, in.	0.017
Material	Zr-4
Instrument thimble	
Number	1
O.D., in.	0.533
Thickness, in.	0.017
Material	Zr-4
U-235 loading	
g/axial cm of assembly @ 4.3 w/o enrichment	54.33
g/axial cm of assembly @ 3.5 w/o enrichment	44.22
g/axial cm of assembly @ 1.5 w/o enrichment	18.95

TABLE 5-2

EQUIVALENT REACTIVITY EXPOSURE

15 X 15 WESTINGHOUSE OFA FUEL

Initial Enrichment, w/o U-235	Equivalent Burnup, GWD/MTU	
	Type A Rack	Type B Rack
1.5	0.0	0.0
2.5	15.83	16.07
3.5	28.21	28.50
4.3	37.14	37.57

Fuel Assemblies Depleted Under Hot Full Power Reactor Conditions and Modeled in the IP3 Spent Fuel Storage Racks, Cold with 0 ppm Soluble Boron and No Xenon.

TABLE 5-3

COMPARISON OF FUEL AND RACK DESIGNS

	Fuel Parameters		Fuel Rack Parameters			
Plant	Type	Enrichment w/o U-235	Poison Material	B-10 loading[1]	Spacing (inches)	Structural Material
Indian Pt. Unit 2	15x15 W	3.5	Borated s.s.	.0044	10.94x10.94	Stainless steel
Indian Pt. Unit 3	15x15 W	3.5				
Type A:			"	.0040	12.0X12.0	"
Type B:			"	"	11.75X12.0	"
Crystal River 3	15x15 B&W	3.3	B_4C with Binder	.012	10.5x10.5	"
Turkey Pt. 3&4	15x15 W	4.5	None		13.66x13.66	"
Summer	17x17					
Reg-1	W	4.3	Boraflex	.0256	10.4x10.4	"
Reg-2		2.3	"	.002	10.4x10.19	"
Reg-3		1.42	None	----	10.12x10.12	"

1- Areal density, gms B-10/cm^2

TABLE 5-4

SUMMARY OF BENCHMARK CALCULATIONS

Case	k_{eff} CASMO/PDQ	k_{eff} KENO	KENO Ref.
Indian Point Unit 2	.935	.933	12
Indian Point Unit 3			1,18
Type A Rack	.922	.909	
Type B Rack	.920	.918	
Crystal River 3	.920	.907	13
Turkey Point 3 & 4	.934	.937	14
V.C. Summer			15
R-1	.941	.932	
R-2	.906	.902	
R-3	.923	.917	

TABLE 5-5

VARIATIONS IN k_{eff} DUE TO WORST CASE GEOMETRY AND TEMPERATURE EFFECTS

Rack Type	Nominal k_{eff}	Worst Case, k_{eff}		Δk_{eff}		Calculational Method
		Geom.	Geom. + Temp.	Geom.	Temp.	
A	.909	.928	.929	.019	.001	KENO-III
B	.915	.932	.940	.017	.008	KENO-III
B	.918	----	.926	----.008+	----	KENO-IV

\+ Combined variation in k_{eff} due to worst case geometry and temperature.

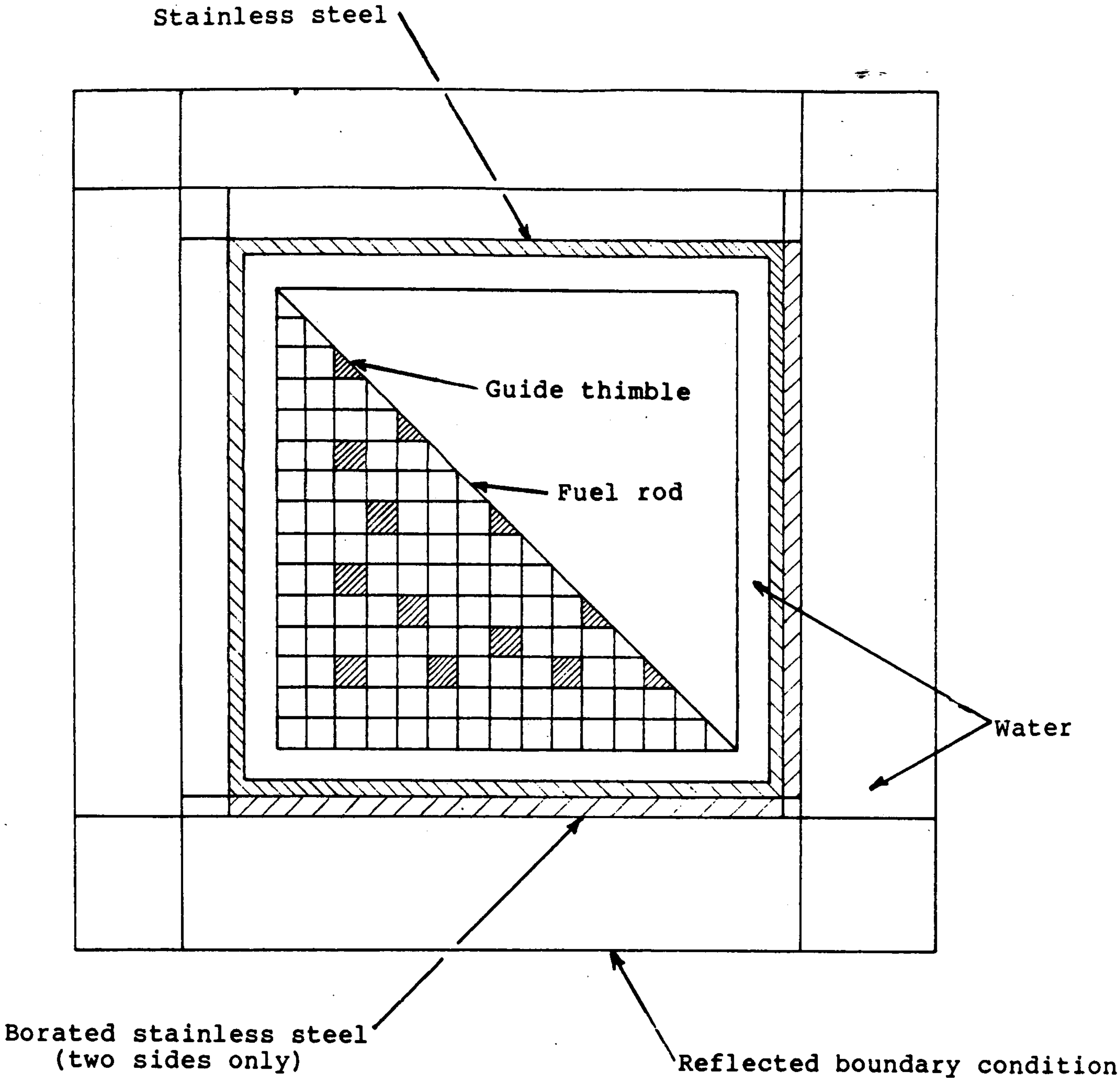


FIGURE 5-1

CASMO-2E FUEL RACK MODEL FOR THE TYPE A STORAGE RACK

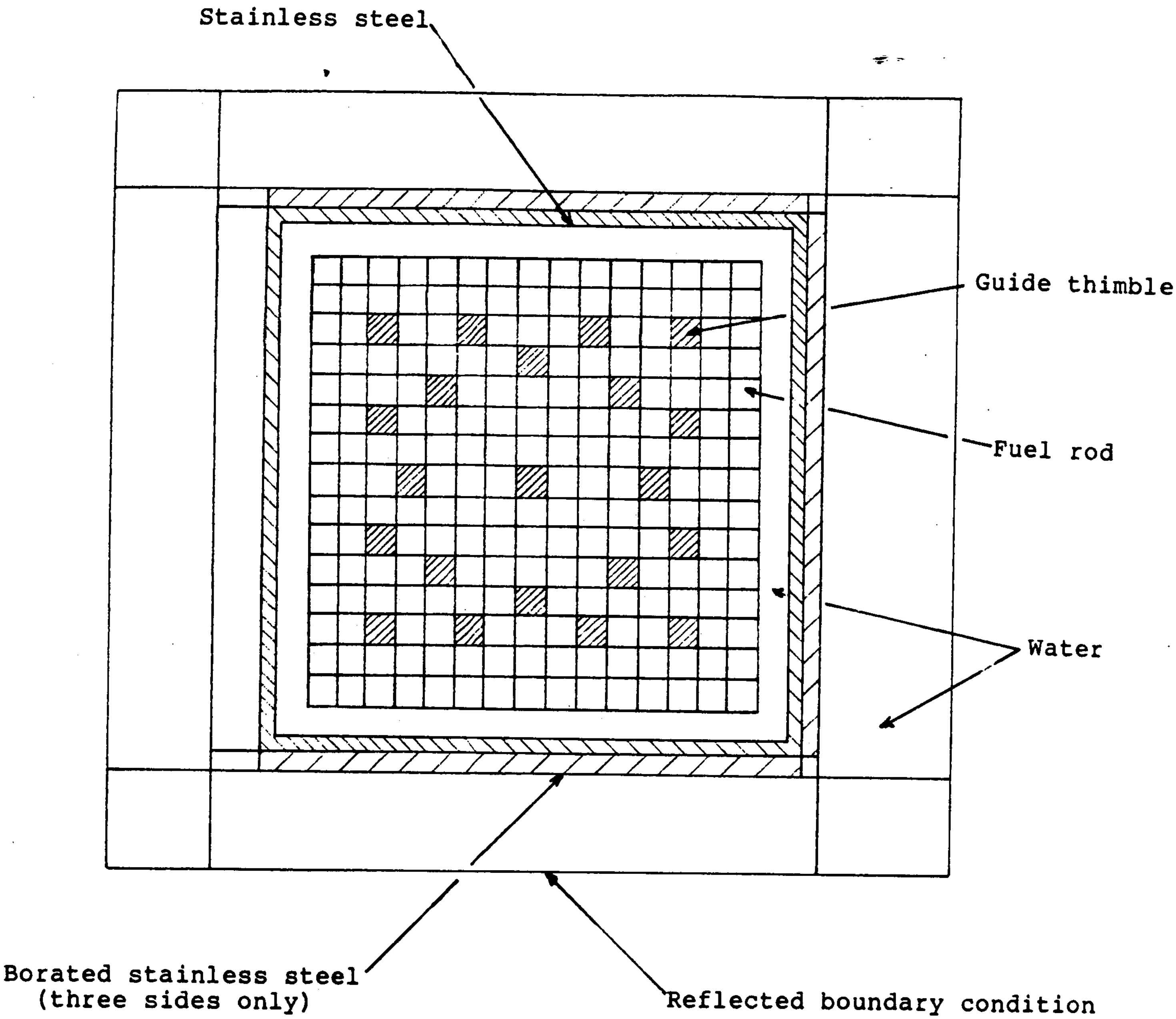


FIGURE 5-2

CASMO-2E FUEL RACK MODEL FOR THE TYPE B STORAGE RACK

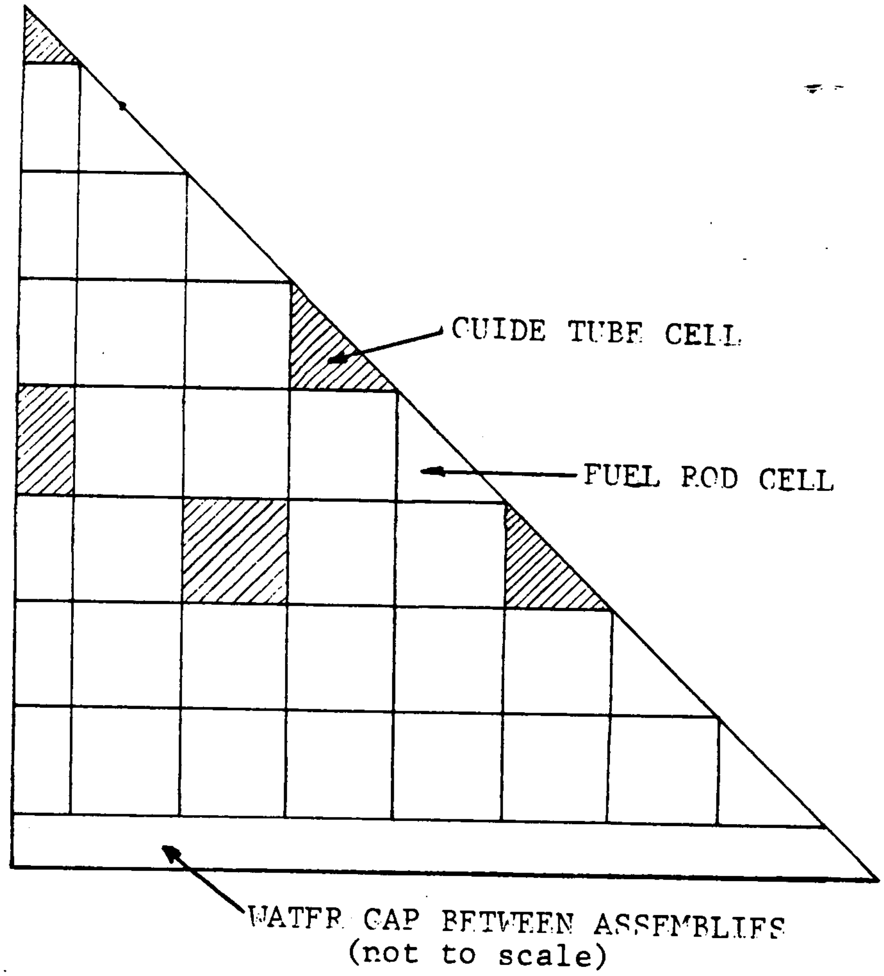


FIGURE 5-3

CASMO 1/8 FUEL ASSEMBLY MODEL

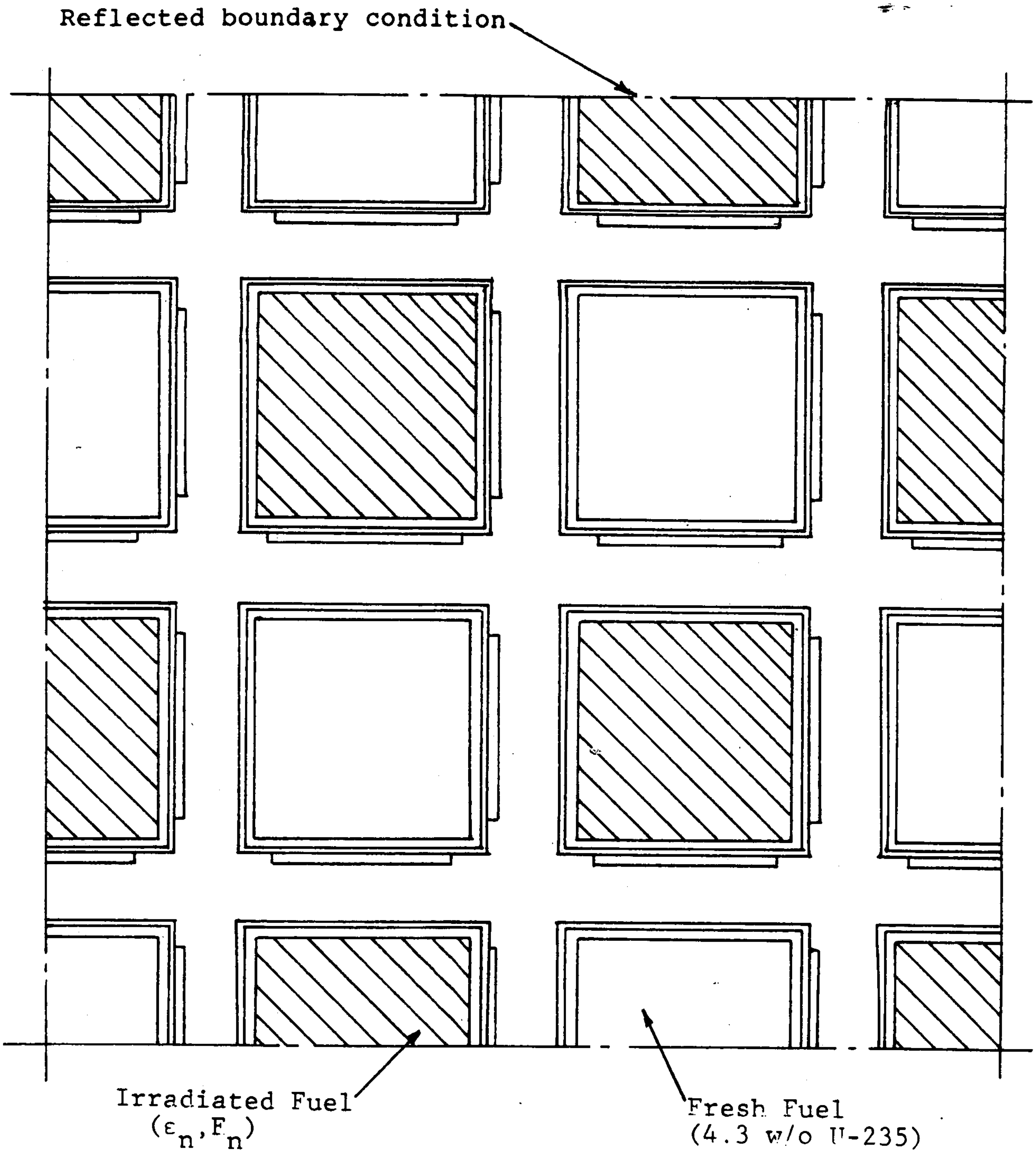


FIGURE 5-4

PDQ MODEL FOR THE TYPE A STORAGE RACK

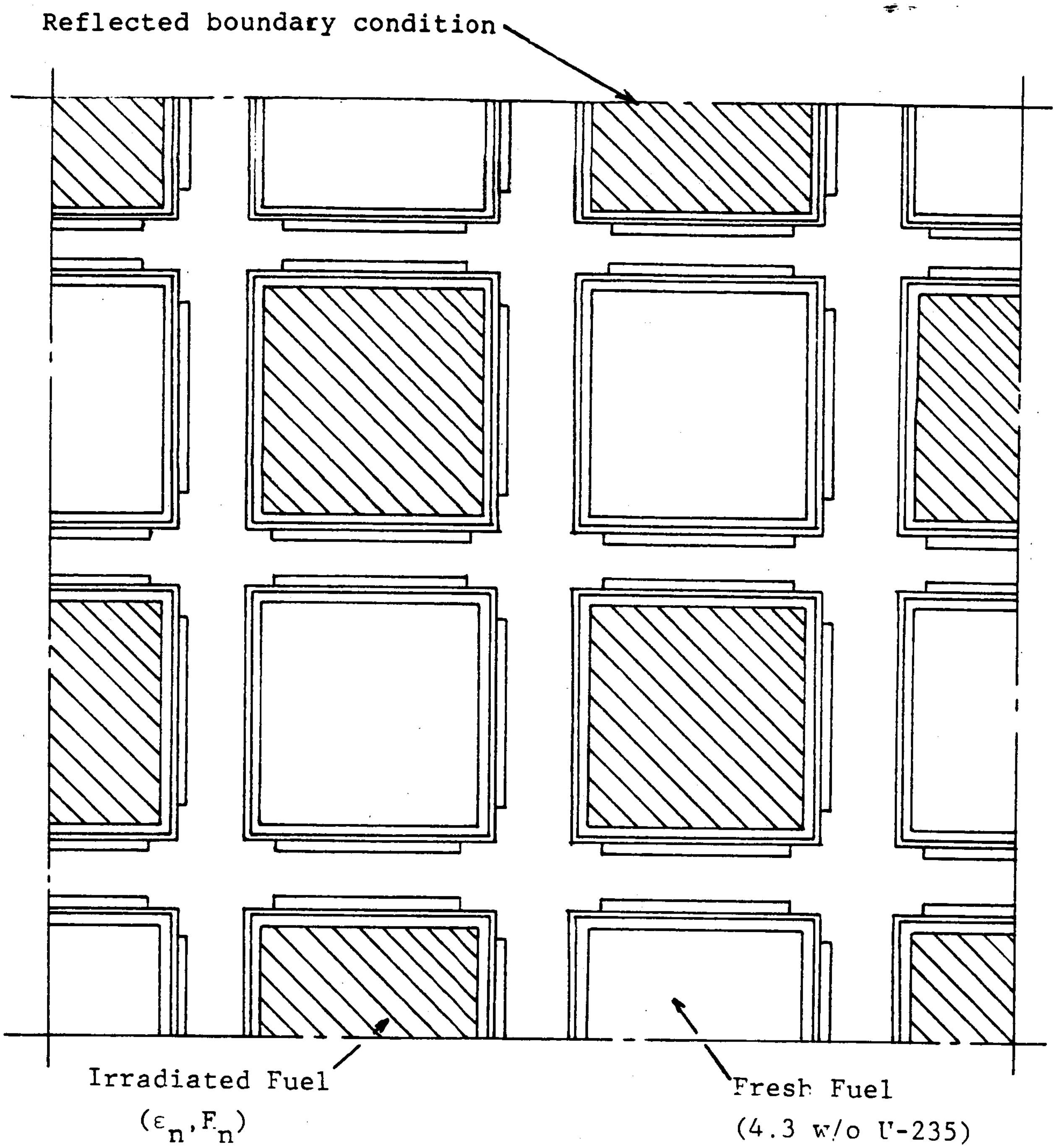


FIGURE 5-5

PDQ MODEL FOR THE TYPE B STORAGE RACK

6. CONCLUSIONS

Nuclear analysis has been completed to demonstrate that the existing spent fuel storage racks at Indian Point 3 can safely store fuel with initial enrichments of up to 4.3 w/o U-235 provided fuel assemblies are loaded in the racks as specified in Section 4.0 of this report. The results of the analysis show that the k_{eff} of the Type A and B fuel/rack are 0.932 and 0.937, repectively, with due allowance for all variations in k_{eff}, model uncertainties, biases and uncertainties in depletion dependent isotopics. This meets the criticality design criterion of k_{eff} <0.95 and is substantially below 1.0. It is therefore concluded that the spent fuel storage racks, when loaded with fuel as specified in Section 4.0, are safe from a criticality standpoint.

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ENCLOSURE 2

CRITICALITY ANALYSIS NEW FUEL STORAGE ASSEMBLY

NEW YORK POWER AUTHORITY

INDIAN POINT 3 NUCLEAR POWER PLANT

DOCKET NO. 50-286

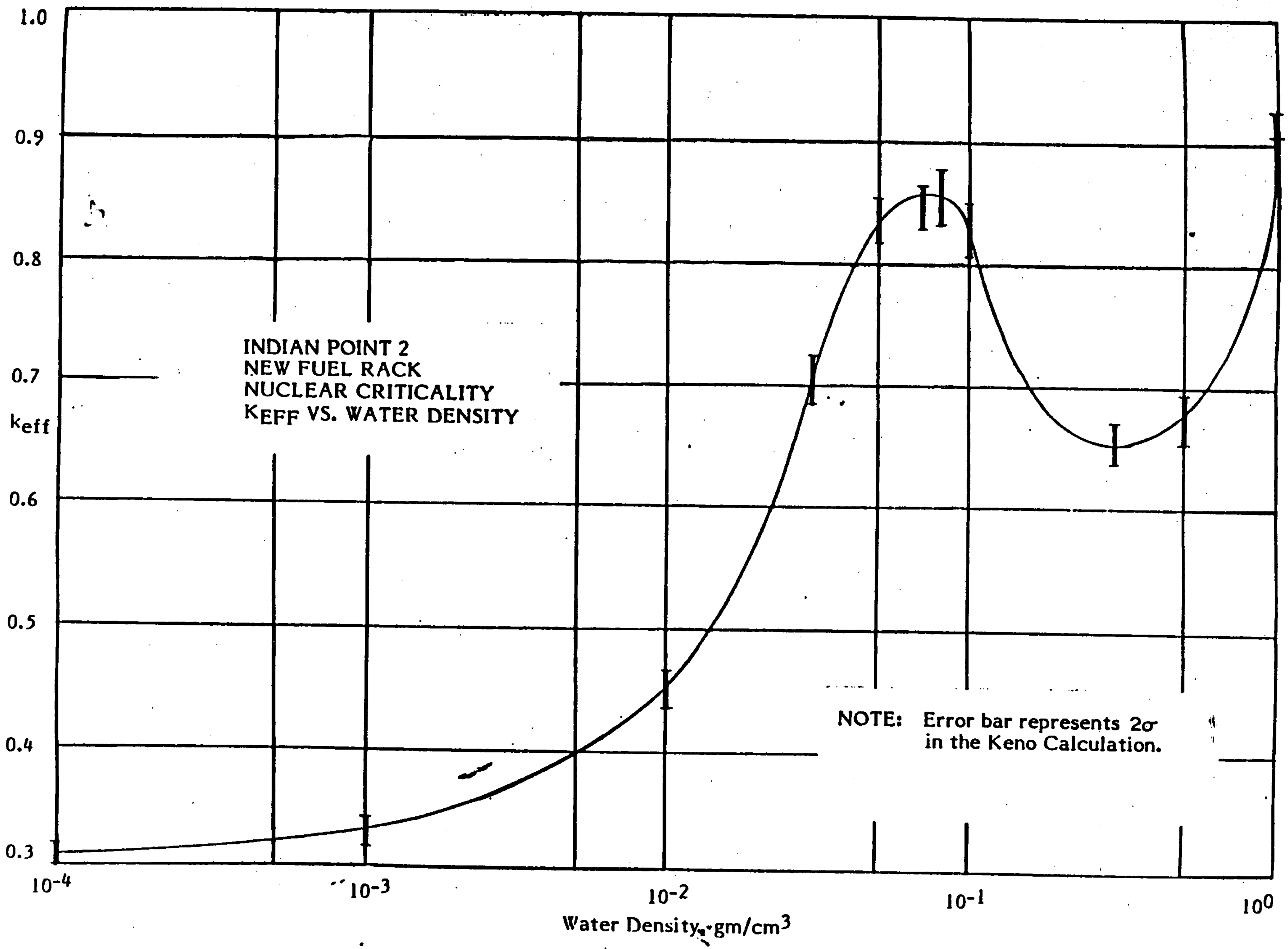
ENCLOSURE 2

CRITICALITY ANALYSIS FOR NEW FUEL RACK

A criticality analysis was performed for the new fuel assembly rack. The calculations included the variation of water density surrounding the fresh fuel assemblies to account for optimum moderating conditions. The results of this analysis are presented in Figure 1.

The Monte Carlo code KENO IV was used for this analysis. The working cross section libraries used as input to KENO IV were prepared from the XSDRN 123 group cross section library, by the NITAWL computer code.

The maximum k_{eff} occurs at maximum water density (.99823 gm/cm^3). This k_{eff}, including all uncertainties and calculational biases, is 0.947 for all normal and abnormal conditions and configurations. This value satisfies the criticality design Criteria of $K_{eff} \leq .95$. Therefore, the new fuel racks meet the criticality design criteria specified in the NRC Standard Review Plan (NUREG-0800) with 15 x 15 Westinghouse fuel assemblies enriched with 4.3 w/o U-235.



INDIAN POINT 2
NEW FUEL RACK
NUCLEAR CRITICALITY
KEFF VS. WATER DENSITY
NOTE: Error bar represents 2σ in the Keno Calculation.
keff
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
10-4
10-3
10-2
10-1
100
Water Density, gm/cm3

FIGURE 1