Docket No. 50-286

ATTACHMENT A

SPENT FUEL POOL MODIFICATION DESCRIPTION, SAFETY ANALYSIS AND ENVIRONMENTAL IMPACT EVALUATION

Power Authority of the State of New York

Indian Point 3 Nuclear Power Plant

July 1977

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1.0 INTRODUCTION

Uncertainties in the future availability of fuel reprocessing facilities have prompted the Power Authority of the State of New York to initiate plans to increase the spent fuel storage capacity of the Indian Point 3 Nuclear Power Plant (IP3). The proposed modification would replace the existing spent fuel storage racks with a new rack design arranged in a more closely spaced lattice array, thereby increasing the spent fuel storage capacity of IP3 from 264 to 840 assemblies. This increased capacity would provide storage space for all spent fuel discharged until 1990 (with full core reserve).

The modification is scheduled to be completed prior to April 1978, the earliest estimated date of the first refueling. It is intended that the necessary support modifications be completed, and that the fuel storage racks be installed while the fuel storage pool is dry. This will permit the activities to be carried out without the additional procedures and safety considerations that would be necessary if there were irradiated fuel stored in the spent fuel pool.

2.0 DESCRIPTION OF PROPOSED MODIFICATIONS

The fuel storage racks will be designed to store the spent fuel and failed fuel canisters designed to be used at IP3. The storage modules will accommodate 837 spent and 3 failed fuel assemblies, as shown in fig. 2.0-1, IP3NPP Fuel Storage Pool Module Arrangement. The center-to-center spacing of the storage locations will be reduced from the present 20.5x20.5 inch lattice array to 12x12 inches in the eastern portion of the pool and to 11¼x12 inches in the western portion of the pool (see fig. 2.0-2 IP3NPP Fuel Cell Arrangement).

Boron-stainless steel plates running the full length of the active fuel region of an assembly are to be welded on to the sides of each storage location between storage positions, as shown in fig. 2.0-2. These boron-stainless steel plates (dim. 145"x7"x1/8") have a minimum boron content of 1% boron. The increases in reactivity caused by the reduction in spacing will be offset by the use of these absorbers.

2.1 Mechanical Design

The fuel storage racks will store spent fuel and failed fuel canisters designed to be used at the Indian Point 3 Nuclear Power Plant (IP3NPP). The spent fuel module construction concept design utilizes a rigidly constructed frame consisting of a top grid, a base frame, vertical columns, and diagonal bracing. The frame supports the vertical spent fuel cells which house the spent fuel. Spacer bars at mid-height of the cells maintain the cell-to-cell spacing during a seismic event.

Overturning of the modules during a seismic event is prevented by fastening the modules together at the top grid with tie bars. Horizontal seismic loads are resisted by a combination of friction and the existing 4½ inch diameter pins on the pool floor. One rack in the northeast corner of the pool will also be supported at the top grid elevation by wall braces to resist overturning. Vertical loads will be transmitted to the pool liner through remotely adjustable screw feet.

2.2 Nuclear Criticality Consideration

The spent and failed fuel racks will be designed so k_{eff} is limited to a value less than 0.95 under all credible circumstances with due consideration to calculational uncertainties. Credit will not be taken for any burnable poisons that may be contained in the fuel, but credit will be taken for neutron absorption of the stainless steel fuel storage cell surrounding each fuel bundle and for the 1% borated stainless steel plates used between fuel cells within a rack module.

The k_{eff} calculations will be based on a maximum fuel enrichment level in new unburned fuel at 3.5% U-235, and will include all credible abnormal fuel configurations. Calculations will consider any reduction in fuel bundle center-to-center spacing resulting from dimensional tolerances, clearance between the fuel bundle and its storage cell, minimum values of boron loadings within the borated stainless steel plates, and deformations under structural loads and from abnormal events. Criticality calculations require the use of four computer codes: the KENO-II Monte Carlo code, to calculate the reactivity of the fuel storage array; the CCELL, BRT-1, and GAMTEC-II codes, to average multigroup cross-section data (18 energy groups); and the XMC Monte Carlo Code, to verify the accuracy of CCELL calculated values of k_{eff} for rod-water lattices.

2.3 Structural Analysis

2.3.1 Loads and Loading Criteria

The loading combinations considered in the fuel rack design will be in accordance with requirements similar to those of the Standard Review Plan Section 3.8.4 of the NRC Structural Design Criteria for Seismic Category I Structures Outside Containment.

2.3.2 Seismic Analysis of Storage Racks

The ground response spectrum curves shown on Figures A.1-1 and A.1-2 of the Indian Point 3 FSAR are to be used in the vertical and horizontal directions, respectively.

A dynamic seismic analysis will be performed to establish seismic loads in the spent fuel racks and associated hardware. Analyses will be performed for the OBE and the SSE conditions. Dynamic seismic analysis will consider two orthogonal horizontal earthquake components and the vertical components and will be applied per Regulatory Guide 1.92.

Linear elastic seismic analysis will be performed by the modal response method using the SAP-4 computer program, a general purpose finite element program. A three dimensional model of a complete rack module will be prepared. The model will include the effects of adjacent rack interactions which would result from the proposed interconnection of the racks. The effects of the water in the pool on the response of the structure will be considered.

Time history non-linear analysis of a single fuel storage cell/ fuel assembly unit will be used to determine the effects of the gaps which exist between store fuel assemblies and their respective storage cells. The analytical model and techniques are similar to those utilized in licensing the Arkansas Nuclear One, Unit 1, replacement spent fuel storage racks (Docket 50-313). Analyses will be performed using: (1) a detailed non-linear model explicitly including the clearance gap between the storage cell well and the fuel assembly; and (2) a simplified linear elastic model with no gap between the storage cell walls and fuel assembly. Peak reaction forces under similar conditions for both analyses will be compared.

Analyses will be performed using the ANSYS computer program or a similar program. The non-linear model to be used will be based on typical fuel assembly mass, stiffness and damping values. Friction effects between the lower end fitting of the fuel assembly and the fuel support plate of the storage cell will be accounted for, as well as structural and hydrodynamic drag damping.

2.3.3 Structural Adequacy of Fuel Pool

The reaction loads in the pool liner and concrete under static and dynamic load conditions, which have been described in Section 2.3.1, will be evaluated to assure that the additional pool loading does not exceed the design capability of the structure.

2.4 Cooling Analysis

2.4.1 Fuel Assembly Cooling

The design of the fuel storage racks will provide for removal of heat generated in the stored fuel by natural convection cooling. The design will prevent local boiling in any fuel storage position when the pool temperature is in the normal operating range, based on worst case fuel assembly placement within the storage array. Fuel clad temperatures will always be maintained at values which will not compromise the integrity of the clad.

Maximum heat generation rates per fuel assembly will be based on NRC Branch Technical Position APCSB 9-2 and/or ANSI 5.1. Maximum local fuel rod heat generation rates will be obtained by subsequently applying appropriate peaking factors. Fuel assembly inlet temperature will be set at the maximum for normal operating conditions. Under the assume d loss of forced coolant circulation conditions, the inlet temperature will be taken as local saturation temperature at the top of the fuel storage racks. Flow paths and flow resistances under normal and loss of forced circulation conditions will be identified on a worst case basis.

Simultaneous solutions of the continuity, momentum, and energy equations using Exxon's XCOB 3C/JUL 75B computer code and subsequent hand calculations will provide local and maximum spent fuel clad temperatures, coolant temperature rise, mass flow rates, local pressure, and pressure drops within the fuel assembly; for two-phase flow situations, void fraction and quality will also be computed. Consideration will also be given to the effects on adequate cooling of the fuel drop accident case wherein a fuel assembly will be assumed to lie horizontally across the top of the spent fuel racks.

Maximum spent fuel pool heat loads for the normal and abnormal cases will be calculated using NRC Branch Technical Position APCSB9-2. The normal case consists of discharging 64 fuel assemblies into the spent fuel pool every 15 months until all storage locations are occupied. The abnormal case consists of a discharge of full core at end-of-life into the pool with the remainder of the pool filled according to normal cycling of spent fuel. The worst case is similar to the abnormal case except that the full core discharge will be made at a time before end-of-life to yield the maximum heat load. Cooling time prior to removal to the pool of each reload will be 100 hours for the normal case, and of the core unload 400 hours for the abnormal case.

Based on calculated heat loads the transient temperature versus time, or pool heating rate characteristics, will be defined under the assumed conditions of a failed heat removal system.

2.4.2 Cooling Systems Performance

The heat removal capability of the spent fuel cooling system has been calculated as a function of the spent fuel pool water temperature and is presented in Figure 2.4-1. The analysis is based on FSAR values of 88.2°F for the component cooling water temperature and 1.4 million pounds per hour for the component cooling water flow rate through the spent fuel heat exchanger. With maximum heat loads, the maximum pool water temperature for the normal and abnormal cases will be calculated.

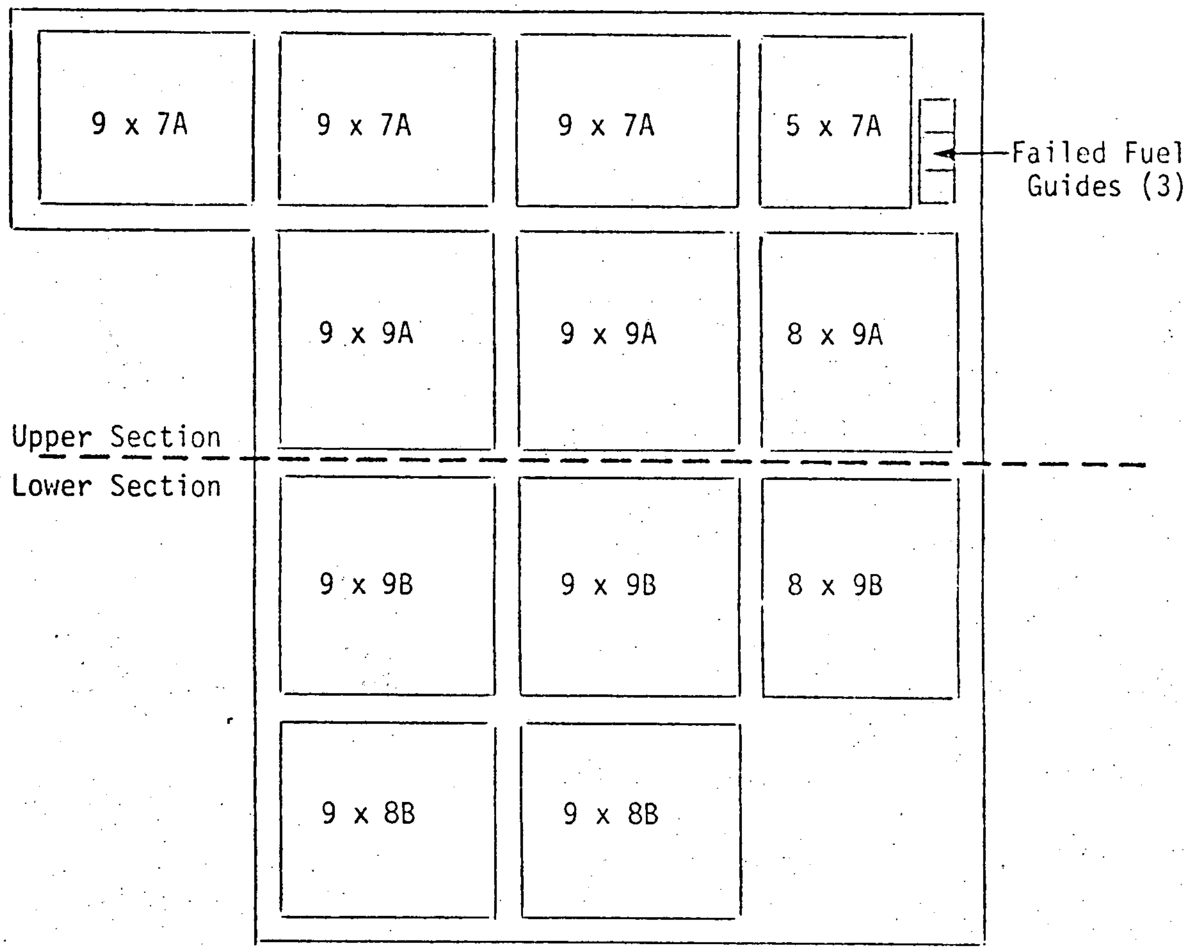
2.5 Quality Assurance

The Power Authority of the State of New York will provide adequate controls for all the activities relative to the proposed modifications by means of a Quality Assurance Program. This program will be identical to the one proposed to and accepted by NRC for the J.A. FitzPatrick Nuclear Power Plant, Docket No. 50-333.

FIGURE 2.0-1

IP3NPP FUEL STORAGE POOL

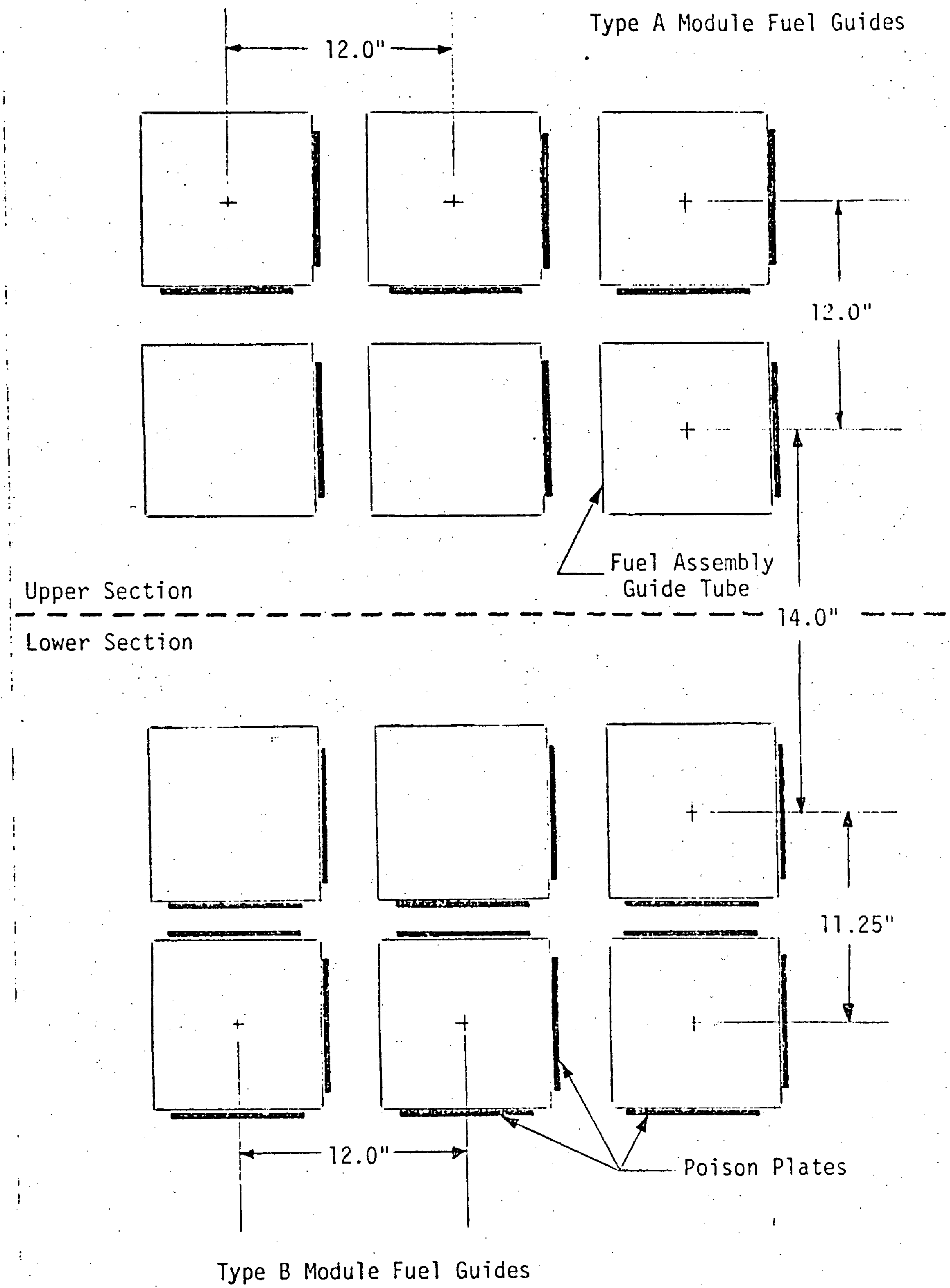
MODULE ARRANGEMENT (TYPES A & B)



Fuel Storage Locations	837
Failed Fuel Storage Locations	3
TOTAL	840

FIGURE 2.0-2

IP3NPP FUEL CELL ARRANGEMENT



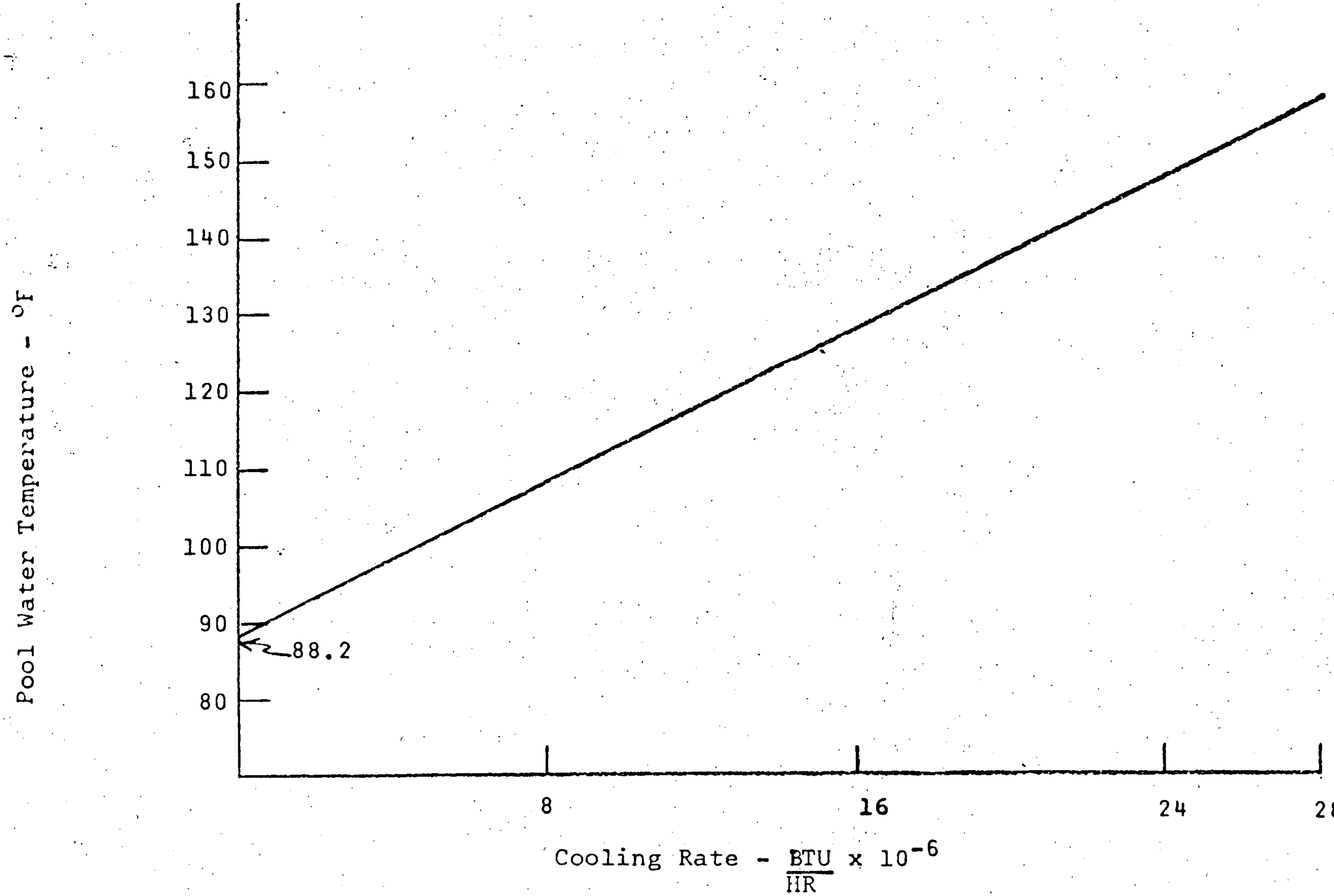


Figure 2.4-1- Spent Fuel Cooling System Heat Removal Capability

3.0 SAFETY ANALYSIS

3.1 Dose Rate Calculations

A radiological analysis will be performed to determine the dose rate to working personnel above and around the spent fuel storage pool. Radionuclide concentrations in the pool will be computed based on information contained in Table 9.2.5 of the Indian Point 3 FSAR.

The radiation transport analysis will be performed using United Engineers & Constructors Inc's HDOSE Model, and shielding calculations will be made with the SHIELD Model.

3.2 Accident Analysis

In addition to evaluating the proposed modification with respect to criticality and cooling considerations, postulated accidents involving spent fuel have been reviewed.

The Indian Point 3 FSAR Section 14.2.1 describes an analysis of four fuel-handling accidents:

a) a fuel assembly becomes stuck inside the reactor vessel

b) a fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pit

c) a fuel assembly becomes stuck in the penetration valve

d) a fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck

Accidents (a), (c) and (d) are not relevant to the design of the spent fuel racks. The accidental dropping of a fuel assembly into the spent fuel pit should not be different in the case of the proposed spent fuel racks from that reported in Section 14.2.1 of the FSAR.

The IP3 Safety Evaluation Report Section 15.3 considered the case of a fuel assembly dropped into the pool with the assumption that all fuel rods of that assembly were damaged. Calculated doses resulting from the release of fission product bases for these accidents were within the guideline values of 10CFR Part 100. The new proposed spent fuel racks in the pool do not affect those analyses, the calculations remain valid and the conclusions applicable.

3.3 Safety Evaluation

For the case of a single region discharge, the existing waiting time requirement of 100 hours would assure that the pool water temperature is well below the design objective. For a full-core discharge, the added requirement of 400 hours total waiting time is expected to limit the decay heat generation rate in the spent fuel pool so that the pool water temperature will not exceed the FSAR design objective.

The spent fuel pool modification will not represent a significant hazards consideration and will not cause any change in the types, or increase in the amounts of effluents.

4.0 ENVIRONMENTAL IMPACT EVALUATION

4.1 Environmental Effects of Increased Storage

4.1.1 Heat Dissipation

The maximum incremental heat load which will result from the accumulation of fuel through 1990 is estimated not to exceed 10^6 Btu/hr. This additional load is insignificant when compared to the station heat rejection rate of almost 5×10^9 Btu/hr at 100% thermal power output.

4.1.2 Radiological Effects

The offsite dose as a result of releases from the spent fuel building is expected to be only a small fraction of total plant tritium release. There will be an increase in the dose rates above the pool due to personnel exposure to the increased concentration of radionuclides in the pool water and the higher density geometry of spent fuel in the pool. A conservative estimate of the pool's activity filled to capacity considers, at least: leakage of isotopes from fuel to pool, decontamination factors and flow rate of the pool purification system, the isotopic half-lives, and the decay time of the fuel. The total dose resulting from routine exposure due to pool enlargement is anticipated to be within the limits of 10CFR20.

4.1.3 Chemical Discharges

No additional chemical discharge to the environment is anticipated as a result of the proposed modifications.

4.2 Environmental Effects of Accidents

The environmental effects of accidents after the proposed modifications of the spent fuel racks are implemented will result in no increase in the environmental impact previously evaluated in the IP3 environmental report.

4.3 Cost Analysis of Alternatives

The total cost associated with the proposed modification is estimated to be under $3 million. Two alternatives to increasing capacity of storage for the IP3 spent fuel pool may be considered for cost comparison purposes, although it is not known that these alternatives would be available:

a. Storage at an independent commercial facility;

b. Storage at a Reprocessing Facility.

ATTACHMENT B

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Power Authority of the State of New York

Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Facility Operating License No. DPR-64

August 1977

Technical Specification changes will be provided with final submittal.