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Lic #DPR-58 & DPR-74 Appl for Amend to increase spent fuel storage capacity from 500 to 2,050 fuel assemblies. W/encl description, safety analysis & environ considerations.

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NEW YORK, N. Y. 10004

November 22, 1978
AEP:NRC:00105

Donald C. Cook Nuclear Plant Units 1 & 2
Docket Nos. 50-315 and 50-316 P
License Nos. DPR-58 and DPR-74
Spent Fuel Storage Capacity Expansion

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

Indiana & Michigan Power Company hereby transmits, attached to this letter, the first part of its application to increase the Donald C. Cook Nuclear Plant spent fuel storage capacity from 500 to 2050 fuel assemblies.

The proposed modification would remove all of the existing spent fuel racks in the spent fuel pool and replace them with high density poison spent fuel racks. The attached application provides a description of the proposed modification, safety analysis and environmental considerations. The information contained herein is presented in a format discussed with your staff on October 31, 1978.

The required structural and thermal analyses will be transmitted to you by December 11, 1978.

Unit 1 of the Donald C. Cook Nuclear Plant is scheduled to refuel in the spring of 1979. Unit 2 is scheduled to refuel in the fall of 1979. It is necessary to perform the proposed modification during a period of time when neither of the two units is refueling since the spent fuel pool is a facility shared by both units. In order to maintain a full core discharge reserve at all times during the spent fuel rack change-out period, we plan to complete this proposed modification before the Unit 2 refueling which is scheduled to commence the first part of October, 1979. Work on this modification is scheduled to commence on August 1, 1979. This schedule will also keep the occupational exposure as low as possible during the replacement period.

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Your immediate attention to this matter is requested.

Very truly yours,

G. P. Maloney
Vice President

Sworn and subscribed to before
this 22 day of November, 1978
in New York County, New York

Kathleen Barry
Notary Public

KATHLEEN BARRY
NOTARY PUBLIC, State of New York
No. 41-4606792
Qualified in Queens County
Certificate filed in New York County
Commission Expires March 30, 1979

cc: R. C. Callen
G. Charnoff
P.W. Steketee
R.J. Vollen
R. Walsh
D. V. Shaller-Bridgman
R. W. Jurgensen

ATTACHMENT 1

INDIANA & MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

DESCRIPTION AND SAFETY ANALYSIS

FOR THE

SPENT FUEL STORAGE CAPACITY EXPANSION PROGRAM

NOVEMBER 1978

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1. Introduction

In March 1978, the Nuclear Regulatory Commission published the Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0404). The findings of that report indicate that the technology of water pool storage is well developed and that the storage of LWR spent fuels in water pools has an insignificant impact on the environment. Also, the physical security measures required for protection against sabotage of stored spent fuel are essentially the same at both reactor and away-from-reactor sites, hence, increased spent fuel storage at either location has little relative safeguards significance. The report also identifies at-reactor compact storage as the most favorable economic cost for providing additional time to develop a program for the final disposal of high level wastes, while allowing for economic and safe nuclear electrical generation.

1.1 History and Need for Replacement

Unit No. 1 of the Donald C. Cook Nuclear Plant achieved initial criticality on January 18, 1975 and Unit No. 2 on March 10, 1978. Unit No. 1 is currently in cycle 3 and Unit No. 2 is in its first cycle. The spent fuel storage pool was designed under the then valid assumption that yearly fuel cycles would be utilized requiring storage of a single batch of spent fuel for less than one year in the spent fuel pool for each unit.

Due to the Government's decision to indefinitely defer the reprocessing of nuclear fuel, we are forced to seek methods to solve the spent fuel storage crisis that is before us. Whatever alternative to reprocessing is chosen, our customers will be affected by higher electricity cost. Replacing our current spent fuel storage racks is, we believe, the safest and most economical means of serving our customers.

1.2 General Description

The spent fuel pool is a facility shared by both Unit No. 1 and Unit No. 2 and is located in the Auxiliary Building between the two Containment Buildings. The general location of the spent fuel pool is shown in Figure 1-1. The present storage capacity is 500 assemblies.

Indiana & Michigan Power Company has entered into a contract with Exxon Nuclear Company, Inc. of Bellevue, Washington for the design, analysis, and fabrication of replacement spent fuel storage racks which will permit the storage of approximately 2050 fuel assemblies in the spent fuel pool. These replacement spent fuel storage racks will provide storage capacity and allow for the continued operation of both Unit No. 1 and Unit No. 2 until approximately the first part of 1992 while still maintaining the capacity for a full core discharge reserve (FCDR) of 193 locations.

The replacement spent fuel storage racks are to be fabricated primarily from type 304 stainless steel. The individual fuel assemblies will be stored in square fuel storage cells fabricated from stainless steel-clad Boral * material. The high density (poison) spent fuel module construction is essentially a replica of the design used in the replacement racks for the Salem Nuclear Generating Station, which the Commission staff is currently reviewing. The module is shown in Figure 1-2.

The design utilizes a stiffened module base and an upper box structure consisting of plate diaphragms and a top grid. The vertical loads are carried by the module base. Horizontal seismic loads are carried to the module base through the plate diaphragms. Tipping is prevented by coupling adjacent racks through a bolted connection at the top grid level.

The detailed design of the spent fuel storage cells is slightly different from the design for the Salem Nuclear Generating Station. Their basic function and construction, however, are similar. Figure 1-3 illustrates the storage cell design for the Donald C. Cook Plant. Each cell is a square cross-section formed from an inner shroud of stainless steel, a center sheet of aluminum clad B_4C, and an outer shroud of stainless steel. This cell acts as a storage space and, in addition, provides sufficient neutron absorption by the boron carbide contained in the Boral sheet to allow spacing of spent fuel in a 10.5 inch by 10.5 inch array. The fuel weight is carried directly on the module base. A flared guide and transition section is provided at the top of each storage cell. This transition is designed to assure ease of entry and to preclude fuel assembly hang-up and damage.

1.3 Specific Needs

Indiana & Michigan Power has a contract with Allied-General Nuclear Services (AGNS) for fuel reprocessing services. Currently, however, no spent fuel can be sent to AGNS for reprocessing due to the December 23, 1977 NRC order terminating licensing proceedings for the Barnwell facility.

Presently, there are 129 spent fuel assemblies stored in the spent fuel pool. Sixty-five assemblies were discharged from Unit No. 1 in January, 1977. The remaining sixty-four assemblies were discharged in April, 1978. One hundred and twelve burnable poison clusters are contained in these assemblies and an additional thirty burnable poison clusters occupy storage locations.

The total storage capacity expected to be utilized is based on maintaining a full core discharge reserve storage capability. The estimated refueling schedules and expected number of fuel assemblies to be transferred into the spent fuel pool are given in Table 1-1. From this table, it can be seen that the existing storage capacity would be filled by May, 1980 with FCDR.

* Trademark of Brooks and Perkins Incorporated

The proposed modifications to the spent fuel pool do not affect the rate of spent fuel generation or the total quantity of spent fuel generated during the anticipated operating lifetime of the facility.

Due to the Government's present position on spent fuel, this proposed expansion will not change the time period that spent fuel assemblies would be stored on-site since we have no place to ship them at this time, and do not foresee shipping them off-site until the early 1990's. The expansion will, however, allow operation of both units for an additional thirteen years without shipping spent fuel off-site.

1.4 Construction Costs

The total cost associated with the project is expected to be approximately $4.7 million. This estimate includes the following items:

1. Project management, design, quality assurance and licensing.
2. Materials, tooling, and hardware fabrication.
3. Removal, installation, transportation, and disposal of the old racks.
4. Contingency allowance.
5. Allowance for funds used during construction.

1.5 Alternatives to Increasing the Storage Capacity

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. In addition, spent fuel storage at an off-site facility is not available at the present time nor is likely to be available before 1981 when our present storage capacity will no longer be adequate.

Shipping spent fuel to another reactor site is not possible since the American Electric Power System has no other nuclear plants. With the present situation in spent fuel storage capacity, we cannot rely on another utility to provide storage space for us.

Furthermore, both the Nuclear Fuel Service's and the General Electric Company's reprocessing plants are in a decommissioned state. Their fuel storage pools are available only in a very limited capacity to a few of their original customers. We do not have access to this storage. The Allied- Gulf Nuclear Service plant is not licensed to operate, and cannot be depended upon for receipt of spent fuel due to the termination of the licensing proceedings for the plant.

There are no independent spent fuel storage facilities available at this time. And due to the uncertainty of licensing such a facility, we do not foresee such a facility as being available in the next ten years. Even if an off-site storage facility were available, we project it to be more economical to store spent fuel on-site and find that there are no environmental benefits associated with off-site storage compared with our proposed action.

If the reactors were unable to refuel due to the existing racks being full, we would be forced to seek replacement of up to 2148 megawatts net electrical energy production. In the short term our system would have to increase utilization of more expensive generation means along with seeking to purchase energy from outside the system. In most instances, we would not be able to purchase this large amount of energy. Then we would have to eliminate or curtail service. This would have a severe adverse socio-economic impact on the customers and communities we serve. If it were possible to purchase this energy, we estimate that the additional cost to our customers would be approximately $1.5 million in today's dollars for each day that the reactors were not operating.

In the long term the energy production would have to be replaced with new generation facilities. Replacement generating units could not be built and placed into service until 1986 at the very earliest. The installed cost of a generating unit to replace the idle Donald C. Cook capacity time is projected to be an investment of more than $2.1 billion.

1.6 Commitment of Material Resources

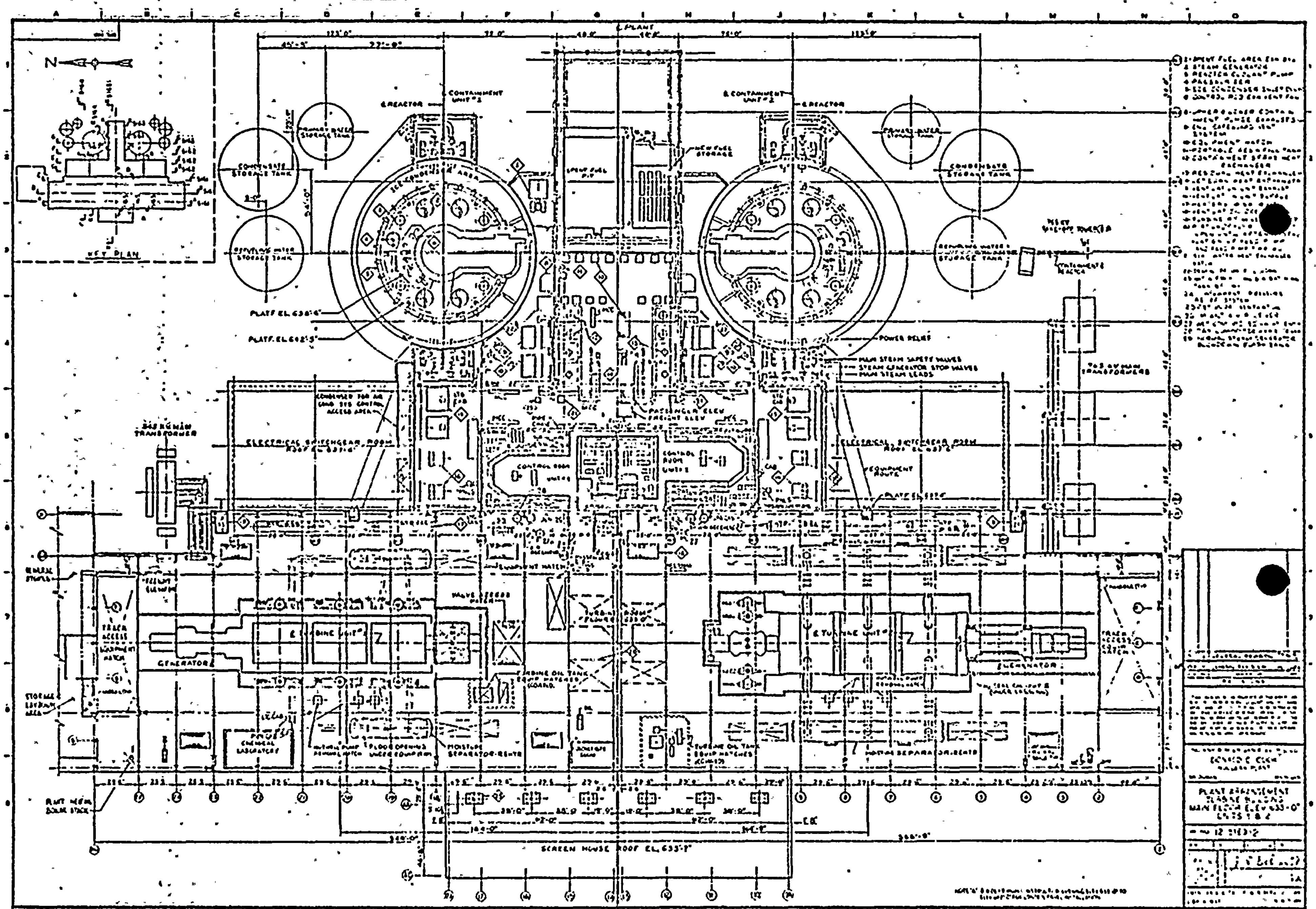
The proposed modification will utilize racks made of stainless steel, boron carbide and aluminum. These materials are readily available in abundant supply.

The material requirements for this one time installation are insignificant compared to the annual national use of these materials and do not represent a significant irreversible commitment of natural resources.

Based on the evaluation of these alternatives and the commitment of resources, we have concluded that increased on-site storage must be provided since there is no place available to ship spent fuel and shutting down the facility would cause grave economic hardship. In addition, in order not to lose FCDR capability, this modification to the spent fuel rack must be completed during the summer of 1979, between the refuelings of the two units.

FIGURE 1-1

LOCATION OF SPENT FUEL POOL



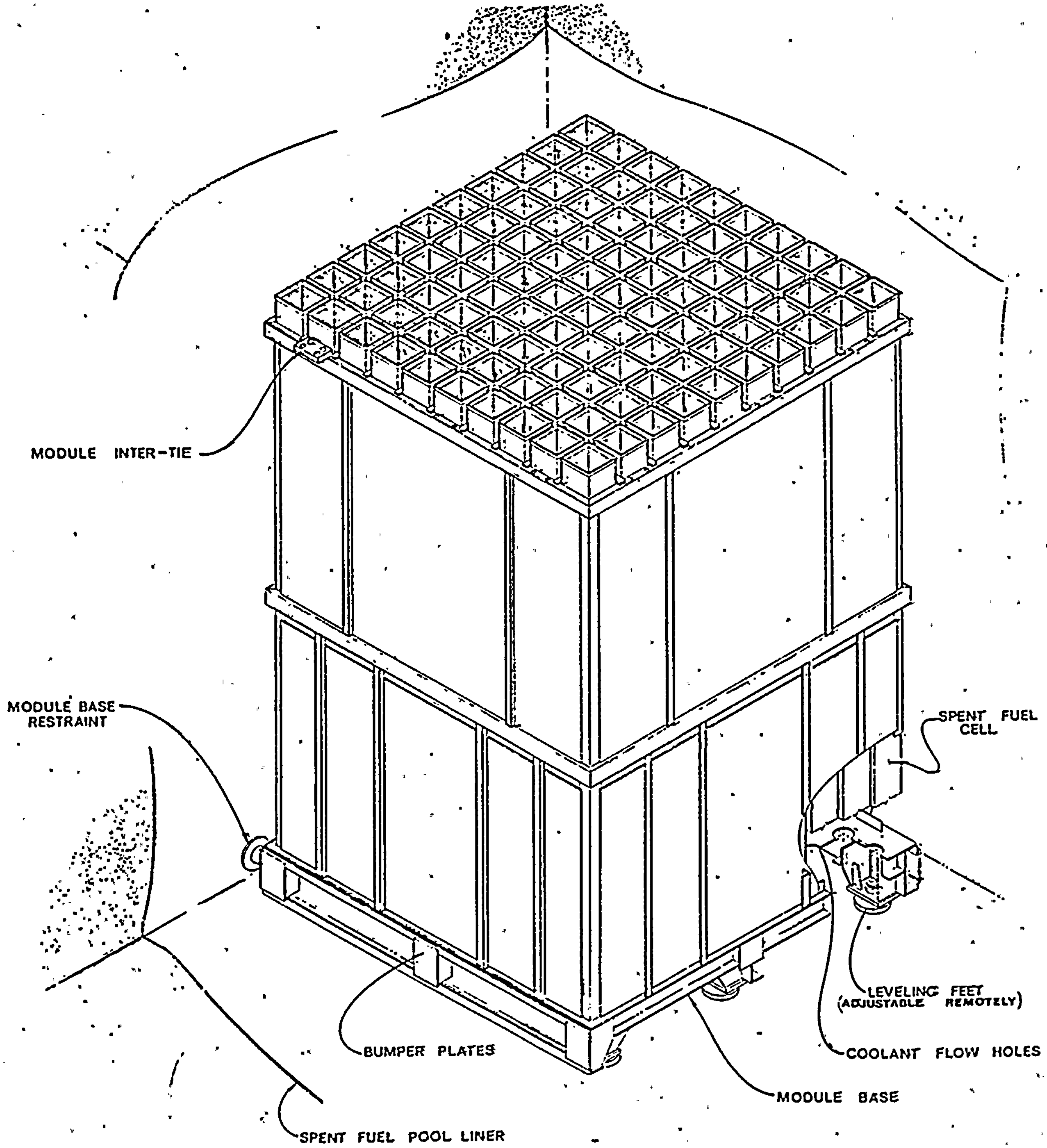


FIGURE 1-2

TYPICAL HIGH DENSITY POISON SPENT FUEL MODULE 10 X 10

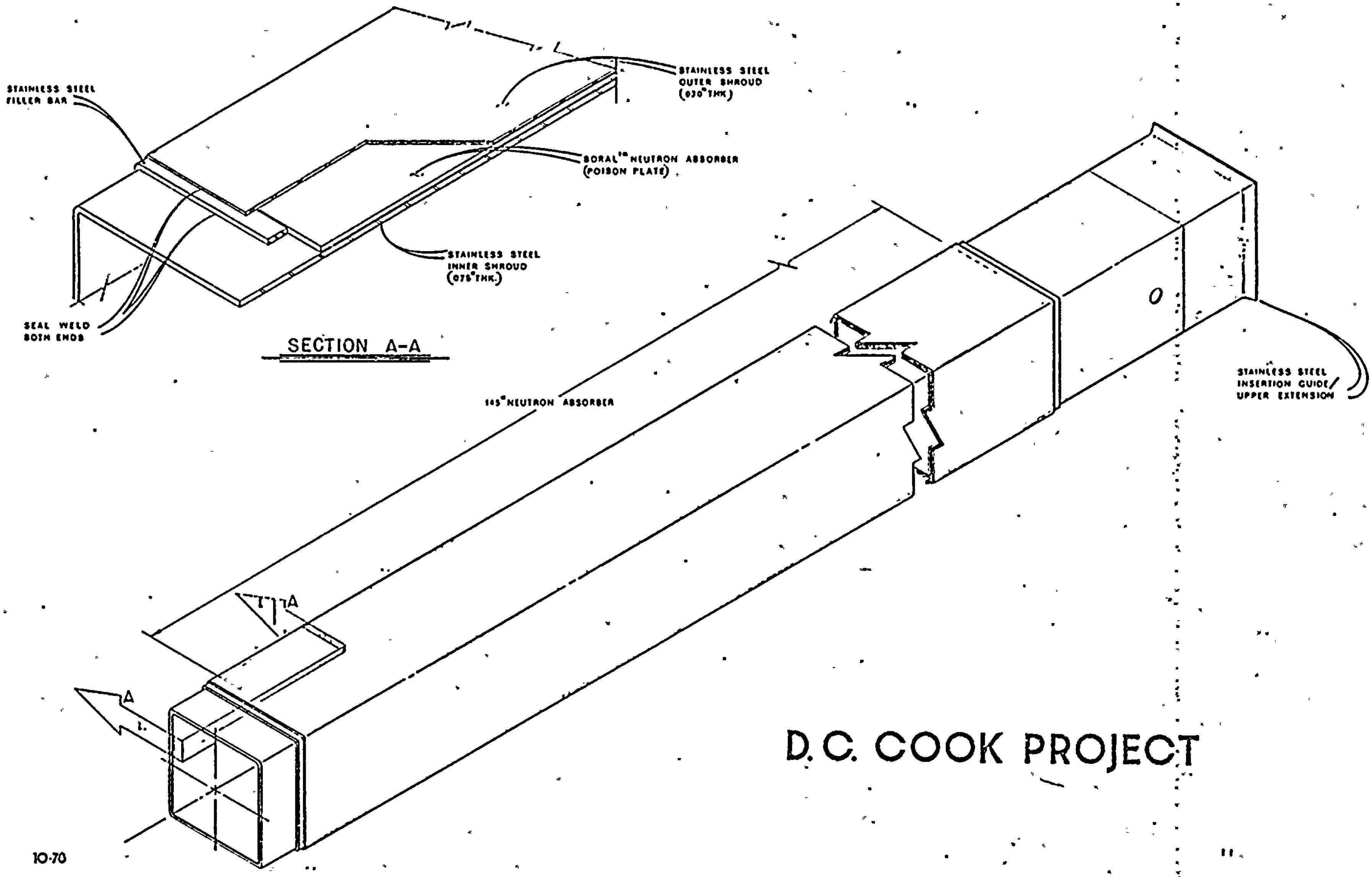


FIGURE 1-3

TYPICAL SPENT FUEL STORAGE CELL

DONALD C. COOK NUCLEAR PLANT

Unit No. 1		Unit No. 2		
Refueling Date	Number of Fuel Assemblies Discharged	Refueling Date	Number of Fuel Assemblies Discharged	Cumulative Number of Fuel Assemblies in SFP
				129
May 1979	64	Oct. 1979	80	273
May 1980	65			338
May 1981	64	Feb. 1981	96	498
May 1982	64	Oct. 1982	84	646
May 1983	65			711
May 1984	64	Feb. 1984	88	863
May 1985	64	Oct. 1985	88	1015
May 1986	65			1080
May 1987	64	Feb. 1987	88	1232
May 1988	64	Oct. 1988	88	1384
May 1989	65			1449
May 1990	64	Feb. 1990	88	1601
May 1991	64	Oct. 1991	88	1753
May 1992	65		Storage Limit ..... with FCDR	1818

TABLE 1-1

2. Radiological Evaluation

We have evaluated the potential radiological impact associated with possible increased releases of solid, liquid and gaseous wastes resulting from the proposed modifications to the spent fuel pit at the Donald C. Cook Nuclear Plant and have found them to be environmentally insignificant. Our findings are discussed below.

2.1 Solid Waste

Installation of additional spent fuel storage capacity in the SFP will require additional reshuffling of the assemblies, which could result in additional crud (corrosion product oxides) being dislodged from the surface. While it is unlikely, storing additional decayed spent fuel could result in some additional fission products being introduced into the SFP water. At present the Donald C. Cook Nuclear Plant Spent Fuel Pit Cooling System (SFPCS) has a maximum system flow rate equal to twice the flow rate required to maintain purity of the pool. The details of the present SFPCS are discussed in Section 9.4 of the Plant's Final Safety Analysis Report. Additional spent fuel could increase the amount of radioactivity accumulated in the filters and demineralizer which are disposed of as solid waste. This increase, if any, should be minor because this modification increases only the storage capacity and not the frequency or the amount of fuel to be replaced for each fuel cycle. The amount of corrosion products released into the pool during any year would be essentially the same regardless of storage capacity.

At present we replace the filters in the purification and skimmer systems about six or seven times a year. The demineralizer resins have not required changing to date. The resin and filter components are changed on the basis of contact radiation levels, pressure drop, chemical exhaustion or breakthrough. We do not expect any increase in the amount of solid waste generated from the spent fuel pool cleanup system due to the proposed modification.

The present spent fuel racks in the SFP will be disposed of as low activity waste. The volume of the racks would be approximately 11,475 cubic feet. This disposal will occur once in the lifetime of the plant. Averaged over the lifetime of the plant, this would increase the total waste volume shipped from the facility by less than 19 %. This would not have any significant additional environmental impact.

2.2 Liquid Waste

Normal operation of the SFP purification system generates some radioactive liquid wastes which are processed by the plant radwaste treatment system. In any case since the amount of radioactive liquid waste generated by the storage facility fuel is minor compared to the total volume of liquid waste generated by plant operations, the proposed modification should add no significant amount of liquid wastes.

The effect of crud buildup on the wall was investigated by taking dose rate measurements at the middle and at the edges of the pool. No significant difference was observed in the readings. Based on this, it is evident that the crud buildup on the wall has little or no contribution to the radiation levels.

Pool leakage is handled by the Spent Fuel leak protection system. Any leaking water is routed to the 587' level of the Auxiliary Building so that observations can be made to determine any leaks. Water is channeled to the Auxiliary Building Sump, then to the radioactive liquid waste hold up tank, and finally, to the waste evaporator. Acceptable liquids are released and the solids are sent off-site for disposal. This modification should not increase any liquid wastes generated by the leak protection system.

A table showing most recent gamma isotopic analysis of SFP water identifying the principal radionuclides and their respective concentrations is shown below:

Isotope	μ Ci/ml
Co-57	2.24 E-5
Cs-134	1.94 E-3
Cs-137	2.63 E-3
Cs-58	1.70 E-3
Mn-54	8.65 E-5
Co-60	2.47 E-3

2.3 Gaseous Waste

Data by year, for the last two years, for KR-85 measured from the auxiliary building ventilation system are given below.

1976	last half	10.01	Ci	Kr-85
1977	first half	161.40	Ci	Kr-85
	last half	13.05	Ci	Kr-85
1978	first half	124.40	Ci	Kr-85

Note that the only gaseous release data available are for batch releases which include gas decay tanks and containment releases.

Given below is a table of recent analyses made to determine the principal airborne radionuclides and their respective concentrations in the SFP area.

Isotope	μ Ci/cc
I-131	≤ 9.88 E-15
Xe-135	≤ 1.49 E-8
Xe-133	≤ 5.73 E-8
Kr-85	≤ 7.44 E-6
Kr-85m	≤ 1.71 E-8
Kr-88	≤ 4.89 E-8
Kr-87	≤ 3.99 E-8
Xe-133m	≤ 1.25 E-7

2.4 Normal Operation Dose Rates

In 1978 the filter was changed six times with a total exposure of 0.324 man-rem. An increase in the frequency of filter changes would increase the exposure proportionately. No estimates for exposure during resin replacements are available since none have been performed so far. These will be done remotely so exposures should not be significant.

A recent measurement of dose rate above the pool indicates a reading of 1 to 3 mR/hr at one foot away from the pool surface.

2.5 Occupational Exposure Due to Change-Out of Spent Fuel Racks

We have reviewed our plans for removal, disassembly and off-site shipment of the old racks and installation of the new racks. This operation is expected to last for a short period of time, no more than 20 weeks. The occupational radiation exposure for this operation is estimated to be about 20 Man-rem. We consider this to be a conservative estimate. This exposure is of the same magnitude as those expected from other special maintenance operations which are to occur periodically during the facility lifetime. Since this is a one-time exposure, it is not directly comparable to the annual doses during normal operation of the SFP.

The increment on on-site occupational dose resulting from the proposed increase in spent fuel storage capacity has been estimated by using realistic assumptions for occupancy times and radioactivity of the spent fuel assemblies themselves. This modification should have a negligible effect on the dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification will add less than 7% to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure will not affect our ability to maintain individual occupational doses as low as reasonably achievable and within the limits of 10 CFR 20.

Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

2.6 Nonradiological Effluents

There will be no change in the chemical or biocidal effects from the plant as a result of the proposed modification.

The SFPCS is expected to keep the pool bulk water temperature at or below the design value of 120°F during normal refuelings until the modified pool is filled. A high temperature alarm is located in the Control Room and set at 125°F. Indicators are in the SFP Cooling System Room.

2.7 Impacts on the Community

The new storage racks will be fabricated off-site and shipped to the plant. No environmental impacts on the environs outside the spent fuel storage building are expected during removal of the existing racks and installation of the new racks. The noise impacts generated within this building are expected to be limited to those normally associated with metal working activities. No significant environmental impact on the community is expected to result from the fuel rack conversion or from subsequent operation with the increased storage of spent fuel in the SFP.

3.0 SAFETY ANALYSIS

3.1 Criticality Considerations

An analysis was performed of the potential maximum reactivity of the fuel stored in the proposed fuel assembly storage facility. This analysis considered the minimum possible spacing under normal and earthquake conditions, the maximum fuel enrichment level, the most reactive conditions of fuel density, and the most reactive water temperature. No credit was taken for any boron present in the storage pool water under normal conditions.

3.1.1 Criticality Criteria

The spent fuel storage racks shall be designed such that K_{eff} is limited to a value of less than 0.95 under normal circumstances when the pool is flooded with demineralized water. The calculated value shall be less than 0.95 by a margin sufficient to account for calculational uncertainties. In the analysis credit may be taken for neutron absorption of the stainless steel clad and for the boron carbide contained within the Boral plates used in the poison cells within the rack module. Credit shall not be taken for any burnable poison that may be contained in the fuel. No credit shall be taken for boron dissolved in the spent fuel pool water under normal conditions.

The K_{eff} calculations shall be based on a maximum fuel enrichment level in new unburned fuel of 3.5 w/o U-235. An evaluation of all credible abnormal fuel configuarations shall be made. Criticality calculations shall consider any reductions in fuel bundle center-to-center spacing resulting from dimensional tolerances and clearance between the fuel bundle and its storage cell. The calculation shall also consider variances in boron loadings within the Boral plates and deformations under structural loads and from abnormal events.

3.1.2 Calculational Methods

The methods employed by Exxon Nuclear Company in the criticality safety analysis are the same as those reviewed and approved by the NRC in prior submittals. The following is a summary of that methodology.

The KENO IV Monte Carlo code[1] was utilized to calculate the reactivity of the D. C. Cook Units 1 and 2 storage array. Multi-group cross section data from the XSDRN 123 group data library were generated for input into KENO IV using the NITAWL[2] and XSDRNPM[2] codes. Specifically, the NITAWL code was utilized to obtain cross section data adjusted to account for resonance self-shielding by the Nordheim Integral Method. The XSDRNPM code, a discrete ordinates one-dimensional transport theory code, was then used to prepare spacially cell-weighted cross section data representative of the fuel assembly for input into KENO IV. The XSDRNPM code was also employed to perform several one-dimensional transport k_∞ calculations to establish the relative reactivity sensitivity of the array to boron content in the storage cells.

In addition to these codes, the CCELL[3] code was used to examine the effects of UO_2 pellet density, moderator temperature, fuel temperature, and enrichment on the infinite multiplication factor of the W 17x17 fuel assembly. CCELL is a pin cell calculation code proprietary to Exxon Nuclear used primarily to obtain cell averaged multigroup cross section data for rod-water lattices.

3.1.3 Design Base Fuel Assembly Description

The fuel storage array is designed to accept fuel enriched up to 3.5 wt.% ^{235}U. From the standpoint of fuel assembly size and infinite multiplication factor, the three assembly designs currently in use (W 15x15 or Exxon 15x15 in Unit 1 or W 17x17

in Unit 2; see Tables 3.1-1 and 3.1-2) are very similar. Hence, differences in pool k_{eff} values for storage of different assembly design types are deemed insignificant.

The fuel assembly specifications and the lattice cell parameters for all three fuel types are given in Table 3.1-1. The bundle averaged cell parameters were calculated by including the zirconium associated with the control rods and instrument guide tube in the zirconium clad of each fuel rod. Water associated with each guide and instrument tube was included by increasing the unit cell dimensions (lattice pitch). Such assumptions permit a conservative estimation of the effect on reactivity of the extra zirconium and water within the fuel assembly.

The analysis discussed herein assumes the storage of W 17x17 fuel design at a maximum enrichment of 3.5 wt % ^{235}U for all UO_2 fuel rods.

3.1.4 Storage Array Description

The D. C. Cook Units 1 and 2 spent fuel storage pool will accommodate twenty specially designed storage rack modules. Each rack module contains a specific number of fuel assembly locations (e.g., 110 locations for a 10 x 11 module) and installation calls for a 14.8 inch nominal center-to-center fuel cell separation between adjacent modules.

Individual fuel assembly storage cells will be manufactured out of stainless steel clad BORAL™. Each cell guide will have a nominal inside diameter of 8.969 inches and a minimum wall thickness of 0.194 inches.

The assumed storage cell wall material thicknesses used in the calculation maximize the pool reactivity by minimizing the amount of both poison and water present between adjacent fuel assemblies

in the overmoderated array. Storage cells manufactured to the minimum specified dimensions assure a minimum ^{10}B loading between fuel assemblies of 0.040 g/cm^2, assuming a $^{10}B/B_{nat.}$ weight ratio of 0.180.

From a neutronics standpoint, the arrangement of modules in the storage pool results in an essentially infinite array of fuel assemblies in both the axial and radial directions. The nominal storage position assumes normal conditions where each unit within the effectively infinite storage array is concentric in its respective cell.

In addition to the nominally spaced array, the minimum spacing between fuel assemblies and the minimum water gap between adjacent storage cells has been considered. Specifically, the minimum center-to-center separation between adjacent storage cells will be "gauged" to assure a minimum water gap between cells of 0.953 inches, compared to a nominal water gap of 1.118 inches. The fabrication tolerances will ensure that the worst credible spacing in the pool array occurs as a cluster of four adjacent assemblies with other storage cells being spaced the nominal center-to-center distance from that cluster. This arrangement also assumes that fuel assemblies in the cluster are in contact with the inside of each respective cell.

For the postulated accident condition of a fuel assembly lying horizontally across one or more of the storage modules, criticality safety is maintained through neutron isolation. A fuel assembly lying across the top of the modules would be isolated from other fissile material by greater than 20 inches of water. This separation between fuel assemblies essentially isolates, from a neutronics standpoint, the horizontal assembly from those in the module cells and, hence, there is no significant contribution to the overall reactivity of the array.

3.1.5 Results

3.1.5.1 Fuel Assembly Reactivity Calculations

Values of k_∞ were computed for the D. C. Cook Units 1 and 2 design base fuel assemblies assuming both the nominal and bundle-averaged lattice cell parameters as given in Table 3.1-1. These cases were examined to provide insight as to the reactivity effects of the instrument and control rod guide tubes within the fuel assembly. These results (see Table 3.1-2) indicate a slight increase in k_∞ due to the increased moderator-to-fuel volume ratio inside the fuel assembly.

In order to evaluate the reactivity sensitivity to changes in enrichment, the value of k_∞ was calculated for ^{235}U enrichments up to and including 3.9 w/o. These values (see Table 3.1-3) would indicate an increase of approximately 6 mk per 0.1 w/o increase in the specified range.

To evaluate the potential effects of pool temperature on the reactivity of individual fuel assemblies, values of k_∞ were computed for temperatures ranging from 20°C to 100°C. Calculated values of k_∞ (see Table 3.1-3) indicate that increasing the fuel assembly temperature results in a decrease in k_∞ of approximately 1 mk per 20°C increase.

In addition to examining the potential effects of temperature, the effect of UO_2 density changes was also examined. For this criticality safety analysis, the UO_2 density was assumed to be 94% of theoretical. Since increasing the UO_2 density decreases the thermal utilization factor for the fuel, k_∞ of the assembly decreases with increasing density (see Table 3.1-3).

3.1.5.2 Storage Array Reactivity Calculations

The KENO IV Monte Carlo code was used to compute storage pool reactivities for assumed worst credible conditions. The bundle-averaged fuel assembly parameters are given in Table 3.1-1. Reactivity calculations were performed using an effectively infinite representation of the storage array.

In evaluating the overall reactivity of the "as designed" storage array, assumptions were made with regards to the worst credible conditions (from the standpoint of neutronics) that could exist in the pool. Conditions assumed in the "worst case" reactivity calculations include:

1) 3.5 wt % ^{235}U enriched fresh UO_2 fuel;
2) Bundle-averaged fuel assembly parameters;
3) Minimum water gap thickness of 0.953 inches between adjacent storage cells; this accounts for limits on installation tolerances and storage cell deformation due to design structural loads, possible earthquake disturbances, etc. This represents the worst case geometry for the array.
4) Temperature variances (20-100°C) in the pool water;
5) No soluble boron in the pool water.

For the nominal case reactivity calculations, only assumptions 1, 2 and 5 are utilized, and the pool water temperature is assumed to be 20°C.

Table 3.1-4 lists results of pool reactivity calculations for both the nominal and worst case conditions. All worst case conditions given above are concurrently considered in a single calculation. In addition to these assumptions, the non-credible condition of assuming the fuel assembly to have a fuel-moderator temperature of 20°C and the water between fuel assemblies to be

at 100°C was made. This assumption maximizes both the reactivity of the fuel assembly and the interaction between adjacent assemblies. For this non-credible boundary case, the reactivity was calculated to be 0.923 ± .004.

In evaluating the effect of storage cell boron loading on array reactivity, several reactivity calculations were performed using XSDRNPM. A storage cell and associated fuel assembly were represented for the calculation by a cylinder with a geometric buckling equivalent to that of the actual fuel assembly. The boron density in the infinite array of storage cells was then varied for each individual calculation. Results as described in Table 3.1-5 would indicate an increase in array reactivity of ~0.014 Δk per 0.010 g/cm^2 ^{10}B decrease between fuel assemblies in the specified ranges for this system. To assure the specified minimum boron loading between fuel assemblies of 0.040 g/cm^2 ^{10}B, the actual average loading between assemblies will be greater than the specified minimum.

3.1.6 Systematic Uncertainties and Benchmark Calculations

The calculational methods and computer codes used to assess the criticality safety of the D. C. Cook fuel storage array have been benchmarked against current experimental critical experiments, and the results of these evaluations are discussed below.

To verify the adequacy of our calculational model, a number of theory-experiment comparisons have been made. For the KENO IV(1) reactivity calculations, the cross section data generation and methods of analysis employed in evaluating these critical experiments were the same as previously described to evaluate the reactivity of the Doel 3 storage arrays. Namely, the XSDRNPM(2) and NITAWL(2) codes were used to generate 123 group, cell weighted, resonance self-shielded fuel region cross sections for input into KENO IV.

One set of critical experiments consists of small, water-moderated critical arrays of fuel rods as described by Grob, et al.[5] We have evaluated several of these UO_2 rod-water lattice critical experiments using the KENO[4] codes with 18 group cross section data averaged using the CCELL[3] code and 123 group data averaged as described above. Results of these calculations are shown in Table 3.1-6. It is noted that the KENO calculated reactivities are in good agreement with previously performed DTF-IV[6] transport theory calculations within the statistical uncertainty of the Monte Carlo calculations.

In addition to these correlations, several benchmark calculations have been performed for critical experiments utilizing 4.95 wt.% ^{235}U metal, rod-water lattices[7] and 2.35 wt.% ^{235}U aluminum clad UO_2 rod-water lattices[8]. Results of reactivity calculations for the uranium metal rod experiments are given in Table 3.1-7. Table 3.1-8 lists results for the aluminum clad rod data, recently reported by Bierman, et al.[8] The theory-experiment correlations show that the analytical methods used adequately reproduce the experimental results.

TABLE 3.1-1

DESIGN BASE D. C. COOK UNITS 1 AND 2
FUEL ASSEMBLY PARAMETERS

	Westinghouse 15x15		Exxon 15x15		Westinghouse 17x17	
	Nominal	Bundle-Averaged	Nominal	Bundle-Averaged	Nominal	Bundle-Averaged
Lattice Pitch, in.	0.563	0.5913	0.563	0.5913	0.496	0.5190
Clad OD	0.422	0.4256	0.422	0.4285	0.374	0.3773
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Clad Thickness, in.	0.024	0.0261	0.030	0.033	0.0225	0.0242
UO_2 Pellet OD, in.	0.3659	0.3659	0.3565	0.3565	0.3225	0.3225
Pellet Density, % ρ_T	95	95	94	94	95	94
Enrichment*, wt.% ^{235}U	3.5	3.5	3.5	3.5	3.5	3.5
Active Fuel Rods	204	204	(same as W 15x15)	(same as W 15x15)	264	264
Rod Array	15x15	15x15	"	"	17x17	17x17
Effective Array Dimensions, in.	8.445x8.445	8.445x8.445	"	"	8.432x8.432	8.432x8.432
Control Rod Guide Tube Dimensions (Zr-4), in.	0.545ODx0.017 wall (upper)	N/A	"	"	0.482ODx.016 wall (upper)	N/A
	0.489ODx0.017 wall (lower)	N/A	"	"	0.429ODx.016 wall (lower)	N/A
Instrument Tube Dimensions (Zr-4), in.	0.545ODx0.017 wall	N/A	"	"	0.482ODx0.016 wall	N/A

*Specified enrichment.

TABLE 3.1-2

INFINITE MEDIA MULTIPLICATION FACTORS

Lattice Cell Parameters	Fuel Assembly k_∞ Values (CCELL)		
	Westinghouse 15x15	Exxon 15x15	Westinghouse 17x17
Nominal	1.424	1.425	1.418
Bundle-Averaged	1.435	1.433	1.430

TABLE 3.1-3

W 17 x 17 Fuel Assembly Reactivity Sensitivity (CCELL)

		^{235}U Loading, g/cm ^{235}U, (axial)	k_∞
ENRICHMENT, (w/o):	3.5	44.22	1.430
	3.7	47.25	1.442
	3.9	49.80	1.453
PELLET DENSITY, (% ρ_T): (at 3.5 w/o and 20°C Fuel + Moderator Temperature)			
	90	42.34	1.431
	94	44.22	1.430
	97	45.63	1.429
	100	47.05	1.428
FUEL and MODERATOR TEMPERATURE, (°C): (at 3.5 w/o and 94% ρ_T Pellet Density)			
	20	44.22	1.430
	60	44.22	1.427
	100	44.22	1.423

TABLE 3.1-4

Reactivity Calculations

D. C. Cook Units 1 and 2

Fuel Type: W 17 x 17 (3.5 w/o)

Storage Cell: Stainless Steel Clad BORAL™ - 0.218" total thickness (assumed)
GT ID: 8.986" (assumed)
GT Center-to-Center Spacing: 10.50" (nominal)
^{10}B Loading: 0.020 g/cm^2 per cell plate

Case	Description	$k_{eff} \pm \sigma$ NITAWL-XSDRNPM-KENO IV (123 group)
1	Nominal	0.908 ± .004
2	Worst Case Geometry and Pool Temperature*	0.923 ± .004

*See description of assumed temperature conditions in Section 3.1.5.2.

TABLE 3.1-5

BORON SENSITIVITY REACTIVITY CALCULATIONS

NITAWL-XSDRNPM

Boron Loading*, g/cm^2 ^{10}B	Δk_∞
0.030	+0.016
0.040	0.0
0.050	-0.011

*Between fuel assemblies.

TABLE 3.1-6

CALCULATED K_{EFF} VALUES FOR CYLINDRICAL ROD-WATER CRITICAL LATTICES

(2.70 Wt.% ^{235}U Stainless Steel Clad UO_2 Rods[3])

Square Lattice Spacing, in.	Moderator-to-Fuel Volume Ratio	Exp'tl. Critical Cylinder Radius, cm	CCELL-DTF-IV Calculated Reactivity (k_{eff})	CCELL-KENO II (18-group) Calculated Reactivity (k_{eff})	NITAWL-XSDRNPM-KENO IV (123-group) Calculated Reactivity (k_{eff})
0.435	1.405	26.820	1.016	1.008 ± .006	1.007 ± .005
0.470	1.853	24.294	1.015	1.014 ± .005	1.013 ± .005
0.573	3.357	23.600	1.011	1.003 ± .005	-
0.615	4.078	24.771	1.009	1.010 ± .005	-
0.665	4.984	27.172	1.005	1.005 ± .005	-

TABLE 3.1-7

CALCULATED K_{EFF} VALUES FOR ORNL CRITICAL LATTICES

(4.95 Wt.% ^{235}U Unclad Uranium Metal Rods[7])

Case	Lattice Number	Number of Rods	Critical Water Height Above Lattice, cm	CCELL-KENO II (18-group) $k_{eff} \pm \sigma$	NITAWL-XSDRNPM-KENO IV (123-group) $k_{eff} \pm \sigma$
			Rod-Water Lattice Only		
1	22	203	7.1	0.988 ± .006	-
2	23	195	15.24	0.998 ± .006	0.999 ± .006
			Rod-Water Lattice + U(0.185) Block		
3	104 (Run)	245	9.5	1.001 ± .006	0.993 ± .006
			Rod-Water Lattice + U(0.185) Block + BORAL™ Sheet		
4	105 (Run)	324	15.24	1.038 ± .005	1.000 ± .005
5	-	359	11.94	1.037 ± .006	-

TABLE 3.1-8

CALCULATED K_{EFF} VALUES FOR BIERMAN CRITICAL LATTICES

(2.35 Wt.% ^{235}U Aluminum Clad UO_2 Rods[8])

Case	Experiment Number	Number of Fuel Clusters in Array, Fuel Rods	Critical Separation Between Fuel Clusters, cm	CCELL-KENO II (18-group) $k_{eff} \pm \sigma$	NITAWL-XSDRNPM-KENO IV (123-group) $k_{eff} \pm \sigma$
	Rod-Water Lattice Only				
1	002	1	∞	1.008 ± .005	1.004 ± .005
2	014	2	8.41	1.007 ± .005	0.991 ± .005
	Rod-Water Lattice + 304L Steel				
3	028	3	6.88	0.994 ± .004	0.997 ± .004
4	035	3	11.47	0.997 ± .005	1.000 ± .005
	Rod-Water Lattice + BORAL™				
5	020	3	6.34*	0.995 ± .005	0.999 ± .005
6	016	3	9.03	1.007 ± .005	0.999 ± .004

*6.33 cm assumed in reactivity calculation.

3.2 Fuel Handling Considerations

An analysis of the consequences of a fuel handling accident was performed in the Final Safety Analysis Report for the D. C. Cook Nuclear Plant (Ch. 14.2.1). The Nuclear Regulatory Commission's Safety Evaluation Report for Cook Plant concluded that the analysis was acceptable. The modification proposed for the spent fuel racks would not affect the consequences or probability of that accident, nor introduce a different or more severe accident.

3.3 Cask Drop Consequences

The details of the cask drop protection system are presented in Question 14.15 of the Plant's Final Safety Analysis. The Nuclear Regulatory Commission reviewed this information and ruled to be acceptable. The proposed spent fuel rack modification does not involve the spent fuel shipping cask area. Therefore, the proposed modification does not affect the original cask drop evaluation. Existing Technical Specifications (T.S. 3.9.7) prohibit travel of loads in excess of 2500 lbs over the fuel assemblies while they are stored in the Spent Fuel Pool.

3.4 Material Considerations

All permanent structural material exposed to the spent fuel pool environment that is used in the fabrication of the spent fuel storage racks is 300 series stainless steel mostly 304. This material was chosen for compatibility with the spent fuel pool water.

At the design operating temperature of 120°F, there is no deterioration or corrosion of stainless steel in this environment. There is also no corrosion problem at temperatures up to and including pool boiling. All other structural components in the spent fuel pool system, such as the pool liner, cooling system pipe connections, etc., are made of stainless steel.

The Donald C. Cook high density spent fuel storage cells utilize Boral material sealed between an inner and outer stainless steel shroud. The Boral material will be supplied by Brooks and Perkins, Inc. to Leckenby Company who will fabricate the spent fuel storage module for Exxon Nuclear Company. The stainless steel shroud (or cladding) is Type 304, meeting the requirements of ASME SA240. The Boral consists of an 1100 series aluminum and boron carbide matrix core sandwiched between two layers of 1100 series aluminum cladding. Boron carbide particles act as a neutron absorber. The boron carbide is ASTM-C750-74 Type II or equivalent. Non-destructive testing of the cells will be conducted to insure 100% leak tightness with a 95% confidence level. In addition to these programs, Exxon Nuclear Company will conduct an independent neutron transmission testing program on the completed poison cells.

In summary, the pool liner, rack lattice structure, and cell exteriors are all stainless steel, which has demonstrated good corrosion resistance in PWR spent fuel pool environments. The design, material selection, and the NDE program provide a high degree of assurance that the integrity of the fixed poison material will be maintained. The material used in the new spent fuel storage racks is similar to present components and does not effect or alter previous evaluations.

3.4.1 Poison Verification Programs

Close control and verification of the material properties utilized in the manufacture of the Boral is assured through the manufacturer's Quality Assurance Program and is documented on appropriate material certification reports. Prior to inserting the Boral plates into the finished cell configuration, each plate is identified in order to allow traceability to the end product. Records are generated for each cell identifying the plates installed in that cell by serial number, thereby providing positive assurance that the required plates are in place.

During rack fabrication, additional care is exercised to prevent damage to the stainless steel cladding of the poison cells. Traceability is continued on the cells by providing a cell location map of each fuel storage rack module.

Special handling measures are imposed on the packaging and shipping to minimize the possibility of degrading the quality of the racks during transit. A thorough receipt inspection at the Donald C. Cook Plant is performed to assure no damage has occurred.

Documentation is maintained on all testing and surveillance performed on the poison cells as well as material certification reports on all materials used in the construction of the cell.

3.4.2 In-Pool Surveillance Program

Surveillance specimens are provided to allow for surveillance over the lifetime of the fuel storage racks. The purpose of these specimens is to provide assurance that no unexpected corrosion is occurring which could compromise the integrity of the Boral. The surveillance specimens are in the form of removable stainless steel clad Boral sheets, which are prototypic of the fuel storage cells. These specimens can be routinely removed and examined and then reinstalled in the spent fuel pool.

3.5 Thermal Considerations

3.5.1 Fuel Assembly Heat Removal

The D. C. Cook spent fuel racks utilize stainless steel encapsulated Boral shrouds supported in a stainless steel structural lattice. Adequate flow paths to the fuel assembly inlet are provided by sufficient space beneath the racks and between the racks and the pool walls. A six-inch hole at the bottom of the fuel storage cell serves as the coolant inlet. Flow paths between fuel storage cells within a rack module are provided to remove gamma heating of the inter-cell coolant.

Design Criteria

The high density spent fuel storage rack design provides storage capacity for slightly more than 10-1/2 cores (2,050 spent fuel assembly storage cells). The original fuel storage design provides for storage of 2-2/3 cores. Because of the high density storage (compared to the original design), the design will be reviewed to determine if adequate natural convection cooling is available during normal operation to: (a) maintain fuel rod clad temperatures at acceptable levels; and (b) preclude boiling within the fuel assemblies. Fuel rod clad temperatures will also be evaluated under hypothetical loss of forced coolant circulation conditions where the pool surface is assumed to reach a saturation temperature of 212°F.

3.6 Mechanical Considerations

3.6.1 Design Criteria

Structural design criteria for spent fuel storage racks will be developed to assure conformance with recognized codes and applicable regulatory guides, including the OT Position for Review and Acceptance of Spent Fuel Storage & Handling Applications, April 14, 1978.

3.6.2 Methods of Analysis

The methods employed by Exxon Nuclear Company in the structural design and analysis will be the same as those reviewed and approved by the NRC in prior submittals. The following is a summary of that methodology:

Structural Analysis

The SAP-4[9] computer program is used for static and dynamic analysis of the fuel storage rack structure. The analytical model is sufficiently detailed to allow determination of static and dynamic loads on all rack members and includes rack-to-rack interties and wall braces. Appropriate boundary conditions for the rack interties and the support point nodes at the base of the structure are developed.

The mass of the water enclosed in the spent fuel storage rack is lumped together with the masses of the fuel assembly and the rack structure in the lumped-parameter SAP-4 model. Static analysis output consists of member loads and nodal deflections. Dynamic analysis output includes frequencies, mode shapes, participation factors and member loads.

Static and seismic loads obtained from the SAP-4 model are combined together and with other loads as required by the criteria to calculate stresses in the structural members. The calculated stresses are then compared with the applicable allowable stresses to confirm structural adequacy.

Non-Linear Effects

Time history analysis of a single fuel storage cell/fuel assembly to account for the effects of the clearance gap between the storage cell wall and the fuel assembly will be performed. The

method of analysis will be identical to that submitted by Arkansas Power and Light Company in its letters dated October 18, 1976 and November 11, 1976, and as approved by the NRC in its Safety Evaluation Report for the Arkansas Nuclear One, Unit 1 Spent Fuel Rack Modification dated December 17, 1976.

Dropped Fuel Assembly Accident

An evaluation of the effects of a postulated dropped fuel assembly accident will be performed to confirm that there would be no effect on the spacing of fuel assemblies stored in the racks. The method of design and analysis will be identical to that submitted by Omaha Public Power District in its letter dated June 2, 1976, and as approved by the NRC in its Safety Evaluation Report for the Fort Calhoun Station Unit No. 1 Spent Fuel Rack Modification dated July 2, 1976.

In addition to the analysis for a vertical drop onto the top of the storage racks, the following cases will be evaluated:

a) fuel assembly dropped inside the storage cell.

b) fuel assembly dropped from above the racks but with the assumption that the assembly rotates as it drops and impacts a row of storage cells.

3.7 References

(1) L. M. Petrie and N. F. Cross, "KENO IV: An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory (November 1975).

(2) N. M. Greene, et al, "AMPX - A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL-TM-3706, Oak Ridge National Laboratory (March 1976).

(3) W. W. Porath, "CCELL Users Guide," BNW/JN-86, Pacific Northwest Laboratories (February 1972).

(4) G. E. Whitesides and N. F. Cross, "KENO - A Multigroup Monte Carlo Criticality Program," CTC-5, Union Carbide Corporation Nuclear Division (September 1969).

(5) V. E. Grob, et al, "Multi-Region Reactor Lattice Studies Results of Critical Experiments in Loose Lattices of UO_2 Rods in H_2O," WCAP-1412, Westinghouse Electric Corporation (1960).

(6) K. D. Lathrop, "DTF-IV - A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering," LA-3373, Los Alamos Scientific Laboratory (July 1965).

(7) Information obtained via personal communication with E. B. Johnson and G. E. Whitesides, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September 1976).

(8) S. R. Bierman, E. D. Clayton and B. M. Durst, "Critical Separation Between Subcritical Clusters of 2.35 Wt.% U-235 Enriched UO_2 Rods in Water with Fixed Neutron Poisons," PNL-2438, Pacific Northwest Laboratories (October 1977).

(9) K. J. Bathe, E. L. Wilson, F. E. Peterson, SAP-IV, "A Structural Analysis Program for Static and Dynamic Response of Linear Systems," Earthquake Engineering Research Center Report No. 73-11, Revised April 1976.

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SUBJECT:
FACILITY OPERATING LICENSE 58 AMEND; CHANGE TO TECH SPECS CONCERNING REVISION TO APPENDIX A WITH REGARD TO EXTEND THE HEAT FLUX HOT CHANNEL FACTOR LIMIT TO HIGHER REACTOR EXPOSURES.

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NEW YORK, N. Y. 10004

February 3, 1978

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
DPR No. 58

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Case:

Changes are hereby requested to the Donald C. Cook Nuclear Plant Unit No. 1 Appendix A Technical Specifications in Sections 3.2 and 4.2 (Power Distribution Limits) and the corresponding bases. The purpose of this change is to extend the heat flux hot channel factor limit (F_Q) to higher reactor exposures than is currently provided for in the Technical Specification. The current F_Q limit is applicable for Cycle 2 exposure up to 10,800 MWD/MTU. Planned operation for the balance of Cycle 2 may exceed this exposure. Also the F_Q limit must be revised to be applicable to future cycles.

The proposed Technical Specification changes are included in the Attachment to this letter. These changes are applicable to all future operation of the plant with the currently designed Exxon Nuclear Fuel.

The basis for the proposed F_Q limit is provided in the ECCS analysis for the plant as reported in Reference 1 and Supplements 1 and 2 to this document (References 2 and 3). The NRC staff's evaluation of the Exxon Nuclear ECCS evaluation model applicable to the plant (Reference 4) required changes involving assignment of flow blockage uncertainties during the calculated reflood portion of the LOCA transient. In Reference 2 further changes to the flow blockage values and uncertainties were proposed and justified for application to the plant. In Reference 3 additional information was provided in response to questions raised by the NRC staff regarding Reference 2.

A001 S
1/1

This Technical Specification change request has been reviewed by both the PNSRC and the required membership of the NSDRC, in accordance with the appropriate provisions of our Technical Specification. The results of our review indicate that our request for this Technical Specification change will not jeopardize the health and safety of the plant workers, nor will it in any way compromise the health and safety of the public.

Very truly yours,

John Tillinghast

John Tillinghast
Vice President

JT/mab
Attachments

Sworn and subscribed to before me
this 3rd day of February, 1978 in
New York County, New York

Kathleen Barry

Notary Public

KATHLEEN BARRY
NOTARY PUBLIC, State of New York
No. 41-4606792
Qualified in Queens County
Certificate filed in New York County
Commission Expires March 30, 1979

cc: R. C. Callen
G. Charnoff
R. W. Jurgensen
D. V. Shaller - Bridgman
P. W. Steketee
R. Walsh
R. J. Vollen

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. 58

DOCKET NO. 50-315

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of chanage.

Pages

3/4 2-1
3/4 2-2
3/4 2-4
3/4 2-5
3/4 2-6
3/4 2-7
3/4 2-15
3/4 2-16 (Replaced by pages 3/4 2-16, 3/4 2-17)
3/4 2-17 (Replaced by page 3/4 2-18)
3/4 2-18 (Replaced by page 3/4 2-19)
3/4 2-19 (Change page number to 3/4 2-20)
3/4 3-39
3/4 3-50
B 3/4 2-1
B 3/4 2-2
B 3/4 2-5
B 3/4 2-6

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a +5% target band (flux difference units) about the target flux difference shown on Figure 3.2-4.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the +5% target band about the target flux difference and with THERMAL POWER:

1. Above 75% x T(E) of RATED THERMAL POWER, within 15 minutes:

a) Either restore the indicated AFD to within the target band limits, or

b) Reduce THERMAL POWER to less than 75% x T(E)** of RATED THERMAL POWER.

2. Between 50% and 75% x T(E) of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the +5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2
**T(E) is defined on Figure 3.2-3 and pages 3/4 2-15, 2-16

LIMITING CONDITION FOR OPERATION (Continued)

c) Surveillance testing of the APDMS may be performed pursuant to Specification 4.3.3.6.1 provided the indicated AFD is maintained with the limits of Figure 3.2-1. A total of 6 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

b. THERMAL POWER shall not be increased above 75% x T(E) of RATED THERMAL POWER unless the indicated AFD is within the ± 5% target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

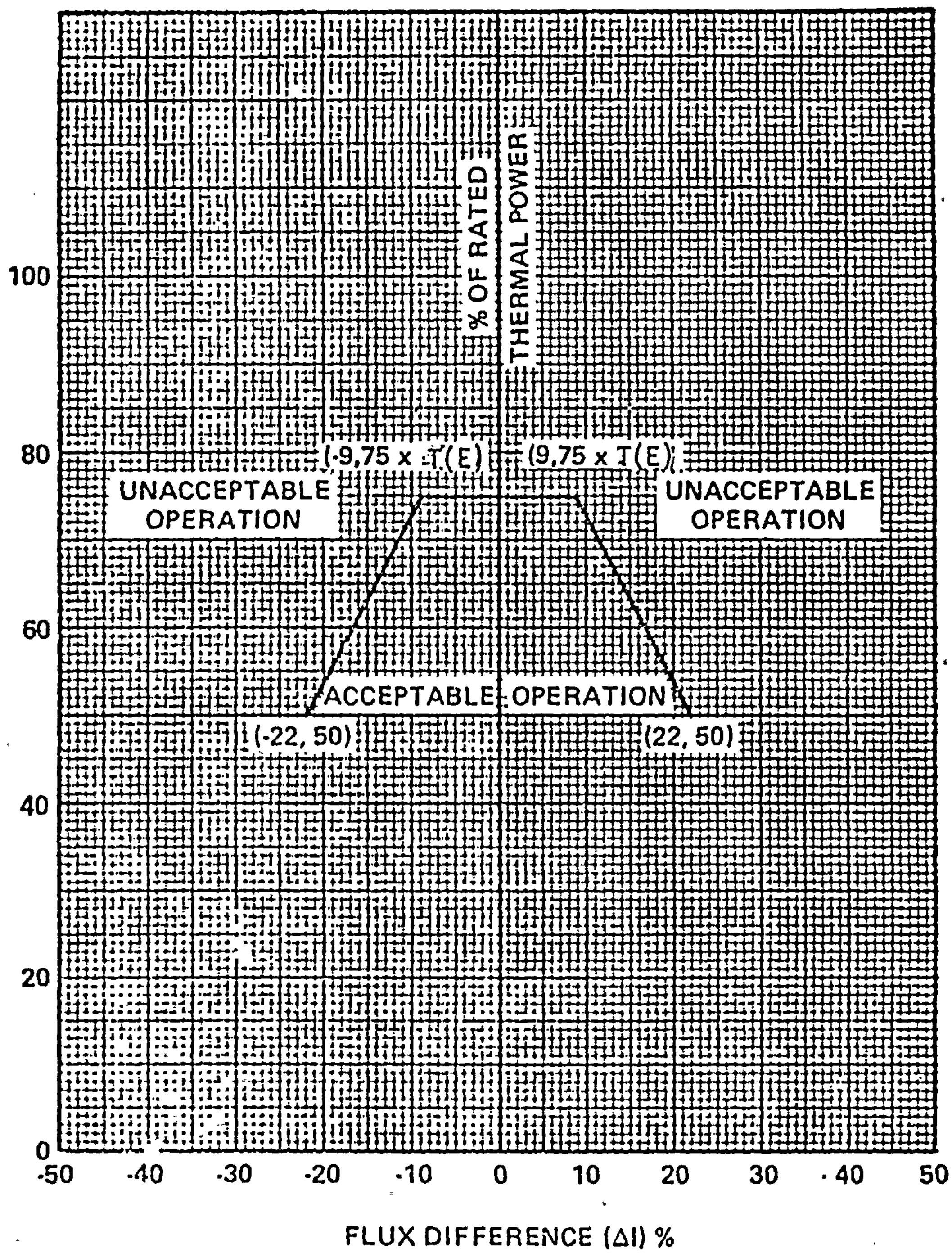

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z,k)$ shall be limited by the following relationships:

$$F_Q(Z,\ell) \leq \frac{[F_Q^L(E_\ell)]}{P} \; [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z,\ell) \leq 2\,[F_Q^L(E_\ell)]\,[K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$F_Q^L(E_\ell)$ is the exposure dependent F_Q limit for assembly ℓ and is defined on Figure 3.2-3 and pages 3/4 2-15, 2-16.
$E\ell$ is the maximum pellet exposure in assembly ℓ.
K(Z) is the function obtained from Figure 3.2-2 for a given core height location.
F_Q is defined as the $F_Q(Z,\ell)$ with the smallest margin or the greatest excess of the limit.

APPLICABILITY: MODE 1

ACTION:

With F_Q exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% F_Q exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor subcritical.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\bar{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided F_Q is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z,\ell)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

2. The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP} \; [1+0.2(1-P)]$$

where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} :

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing either bank "D" control rods or any part length rods, and
2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 18.4 ± 2%, 36.6 ± 2%, 54.7 ± 2% and 72.9 ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" or part length control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}:

1. The $F_Q(Z,\ell)$ limit shall be reudced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}^{L}, and
2. The effects of F_{xy} on $F_Q(Z,\ell)$ shall be evaluated to determine if $F_Q(Z,\ell)$ is within its limit.

4.2.2.3 When $F_Q(Z,\ell)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z,\ell)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[1.95]\,[K(Z)]}{(\overline{R}_{jm'})(P_L)(1.03)(1 + \sigma_{jm'})(1.07)F_P}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_{jm'}$ is the $\overline{R}_{jm}$ for thimble j, and limiting fuel batch m´. $\overline{R}_{jm}$ for each batch m is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns above 84% x T(E) of RATED THERMAL POWER in accordance with:

$$\overline{R}_{jm} = \frac{1}{n} \sum_{i=1}^{n} R_{ijm}.$$

$$R_{ijm} = \frac{F_{Q_{im}}^{Select}}{[F_{ij}(Z)]_{MAX}}$$

$$F_{Q_{im}}^{Select} = [F_{Q_{ik}}^{Meas}/T(E_k)]_{MAX} \text{ by batch m}$$

$F_{Q_{ik}}^{Meas}$ is the relative linear power density for the k^{th} fuel segment in the i^{th} full core flux map.

$$T(E_k) = F_Q^L(E_k)/F_Q^L(0)$$

LIMITING CONDITION FOR OPERATION (Continued)

where

$F^L_Q(E_k)$ = The exposure dependent F_Q limit on the maximum linear heat generation rate for fuel segment k. $F^L_Q(E)$ is defined in Figure 3.2-3.

$F^L_Q(0)$ = The F_Q limit at zero exposure.

The limiting fuel rod in the core is the rod which produces the maximum value of $F^{select}_{Q_{im}}$. The limiting batch m′ is the fuel batch containing limiting fuel segment k′.

Define T(E) as

$T(E) = F^L_Q(E_{k'})/F^L_Q(0)$

$[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of $F^{Meas}_{Q_i}$.

e. $\sigma_{jm'}$ is the standard deviation associated with thimble j, and limiting fuel batch m′, expressed as a fraction or percentage of $\bar{R}_{jm'}$. σ_{jm} for each batch m is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_{jm} = \left[\frac{1}{n-1} \sum_{i=1}^{n} (\bar{R}_{jm} - R_{ijm})^2\right]^{1/2} / \bar{R}_{jm}$$

f. The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system respectively.

The factor 1.03 is the engineering uncertainty factor.

g. F_p is an uncertainty factor to account for the reduction in the $F^L_Q(E)$ curve due to an accumulation of exposure prior to the next flux map. This correction is only required when T(E) for the limiting fuel segment is less than 1.0.

F_p = 1.0 for T(E) = 1.0

F_p = 1.01 for T(E) $\leq$ 1.0

E is the peak pellet exposure in the limiting fuel rod.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODE 1 above 84% x T(E) of RATED THERMAL POWER#.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER one percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next two hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to 84% x T(E) or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by > 4 percent, reduce THERMAL POWER TO 84% x T(E) or less of RATED THERMAL POWER within 15 minutes.

The APDMS may be out of service: 1) when incore maps are being taken as part of the Augmented Startup Test Program or 2) when surveillance for determining power distribution maps is being performed.

SURVEILLANCE REQUIREMENTS

4.2.6.1 $F_j(Z)$ shall be determined to be within its limit by:

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.6 at the following frequencies.

1. At least once per 8 hours, and

2. Immediately and at intervals of 10, 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above 84% x T(E) of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

b. Or using the movable incore detectors at the following frequencies when the APDMS is inoperable:

1. At least once per 8 hours, and

2. At intervals of 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above 84% x T(E) of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

4.2.6.2 When the movable incore detectors are used to monitor $F_j(Z)$, at least 2 thimbles shall be monitored and an $F_j(Z)$ accuracy equivalent to that obtained from the APDMS shall be maintained.

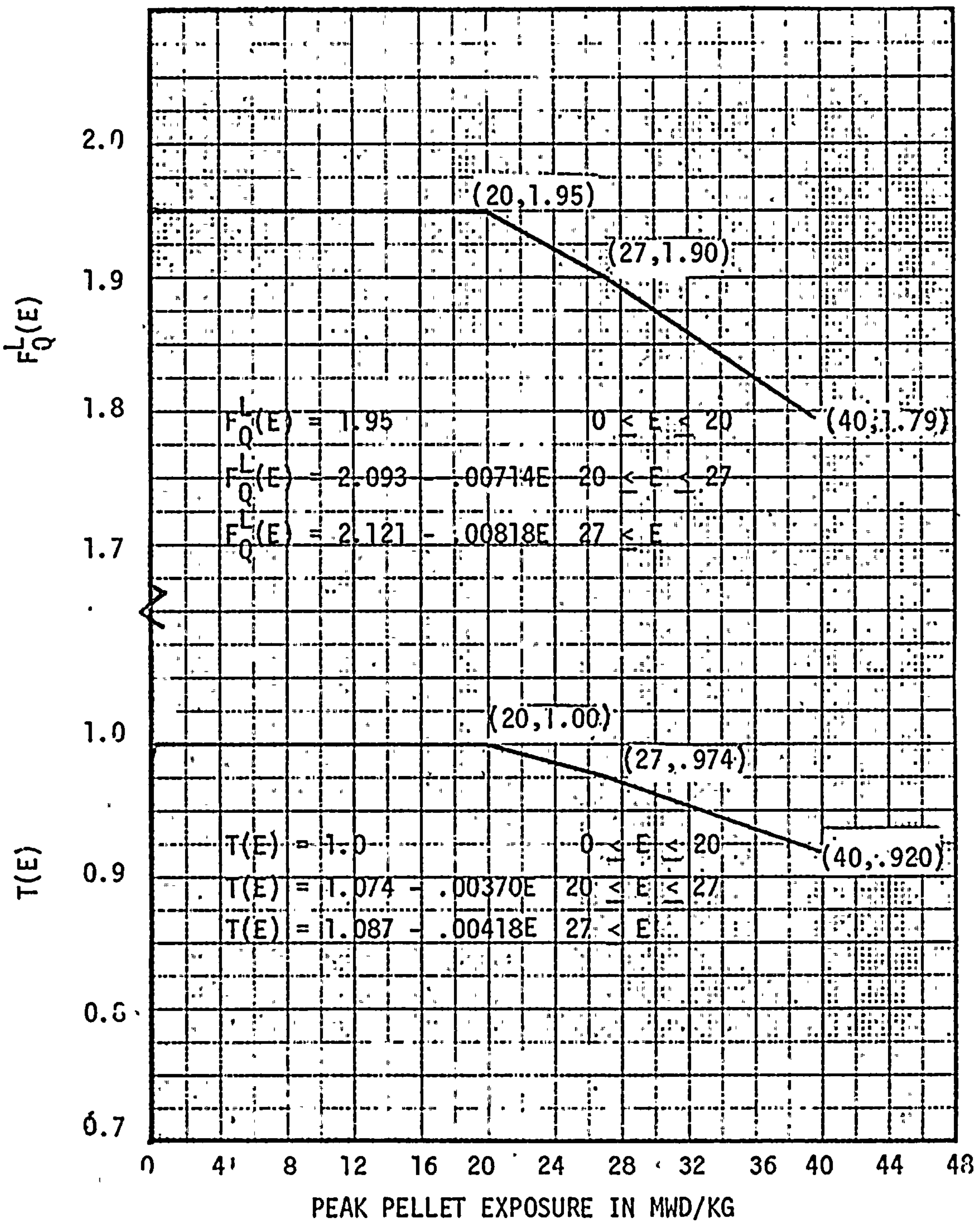

FIGURE 3.2-3 EXPOSURE DEPENDENT F_Q LIMIT, $F_Q^L(E)$, AND NORMALIZED LIMIT T(E) AS A FUNCTION OF PEAK PELLET BURNUP

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the axial flux difference detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z,\ell)$

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output to be used during its use when required for:

a. Recalibration of the excore axial flux difference detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z,\ell)$

SURVEILLANCE REQUIREMENTS (Continued)

a. If the absolute value of $\frac{R_{ijm'} - \overline{R}_{jm'}}{\overline{R}_{jm'}}$ is greater than $2\sigma_{jm'}$ another map shall be completed to verify the new $\overline{R}_{jm'}$. If the second map shows the first to be in error, the first map shall be disregarded. If the second map confirms the new $\overline{R}_{jm'}$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_{jm'}$ and $\sigma_{jm'}$ can be defined from the six new maps.

4.3.3.6.2 The APDMS shall be demonstrated OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to its use and at least once per 31 days thereafter when used for monitoring $F_j(Z)$.

b. At least once per 18 months, during shutdown or below 5% of RATED THERMAL POWER, by performance of a CHANNEL CALIBRATION.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are meet and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z,k)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the + 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 75% x T(E) of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 75% x T(E) of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 75% x T(E) and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

The upper bound limit (84% x T(E) of RATED THERMAL POWER) on AXIAL FLUX DIFFERENCE assures that the $F_Q(Z,\ell)$ envelope of 2.32 times K(Z) x T(E) is not to exceed during either normal operation or in the event of xenon redistribution following power changes. The lower bound limit (50% of RATED THERMAL POWER) is based on the fact that at THERMAL POWER levels below 50% of RATED THERMAL POWER, the average linear heat generation rate is half of its nominal operating value and below that value, perturbations in localized flux distributions cannot affect the results of ECCS or DNBR analyses in a manner which would adversely affect the health and safety of the public.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

BASES

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restriting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

A burnup dependent F_Q is specified as a result of the ECCS evaluation in accordance with 10 CFR Part 50 Appendix K and to meet the acceptance criteria of 10 CFR 50.46. The basis for this dependence is given in document XN-76-51, Supplement 1 and 2.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above 84% x T(E) of RATED THERMAL POWER. This additional limitation on F_Q is necessary in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA.

The unit may operate with fuel assemblies supplied by the Exxon Nuclear Company and by Westinghouse Electric Corporation. The specified limit for F_Q represents the Exxon Nuclear supplied fuel which has the more restrictive power peaking limit.