April 25, 1988

Docket No. 50-289

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Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 67011)

The Commission has issued the enclosed Amendment No.138 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, in response to your letter dated January 12, 1988.

The amendment revises Technical Specification (TS) Sections 5.4.1 and 5.4.2 and associated surveillance requirements to allow receipt, storage and transfer of new fuel assemblies containing as high as 4.3 weight percent U-235 enrichment.

A copy of the related Safety Evaluation is also enclosed. As discussed in our Safety Evaluation, we request that the 12 storage locations required by TS Section 5.4.1.a to be left vacant, be blocked to prevent inadvertent placement of fuel in these locations. The method of blocking is not specified. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/S/

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 138 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDI-4	PM:PDI-4	D:PDI-4	OGC
SNorris	RHernan:bd	JStolz	
04/12/88	04/12/88	04/18/88	04/18/88

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Mr. Henry D. Hukill
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

cc:

G. Broughton
O&M Director, TMI-1
GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057

Richard J. McGoey
Manager, PWR Licensing
GPU Nuclear Corporation
100 Interpace Parkway
Parsippany, New Jersey 70754

C. W. Smyth
TMI-1 Licensing Manager
GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057

Ernest L. Blake, Jr., Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, D.C. 20037

Larry Hochendoner
Dauphin County Commissioner
Dauphin County Courthouse
Front and Market Streets
Harrisburg, Pennsylvania 17120

David D. Maxwell, Chairman
Board of Supervisors
Londonderry Township
RFD#1 - Geyers Church Road
Middletown, Pennsylvania 17057

Richard Conte
Senior Resident Inspector (TMI-1)
U.S.N.R.C.
Post Office Box 311
Middletown, Pennsylvania 17057

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852

Governor's Office of State Planning
and Development
ATTN: Coordinator, Pennsylvania
State Clearinghouse
Post Office Box 1323
Harrisburg, Pennsylvania 17120

Thomas M. Gerusky, Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
Post Office Box 2063
Harrisburg, Pennsylvania 17120

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 138
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated January 12, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.138 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 25, 1988

ATTACHMENT TO LICENSE AMENDMENT NO.138

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
4-1	4-1
4-2	4-2
4-10	4-10
5-6	5-6
5-7	5-7

4. SURVEILLANCE STANDARDS

During Reactor Operational Conditions for which a Limiting Condition for Operation does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/component operable, the associated surveillance requirement must be current. The above applicability requirements assure the operability of systems/components for all Reactor Operating Conditions when required by the Limiting Conditions for Operation.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protection system, engineered safety feature protection system, and heat sink protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Each post accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration at the frequencies shown in Table 4.1-4.

Bases

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protection channels is required monthly on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protection channels is as follows:

Channels A, B, C & D	Before Startup, when shutdown greater than 24 hours
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested monthly. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protection channels and the regulating control rod power SCRs shall be trip tested prior to startup when the reactor has been shutdown for greater than 24 hours.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

TABLE 4.1-3 Cont'd.

	Item	Check	Frequency
4.	Spent Fuel Pool Water Sample	Boron concentration greater than or equal to 600 ppmb	Monthly and after each makeup.
5.	Secondary Coolant System Activity	Isotopic analysis for DOSE EQUIVALENT I-131 concentration	At least once per 72 hours when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F
6.	Boric Acid Mix Tank or Reclaimed Boric Acid Tank	Boron concentration	Twice weekly***
7.	Deleted		
8.	Deleted		
9.	Deleted		
10.	Sodium Hydroxide Tank	Concentration	Quarterly and after each makeup.
11.	Deleted		
12.	Deleted		

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

** Deleted

*** The surveillance of either the Boric Acid Mix Tank or the Reclaimed Boric Acid Tank is not necessary when that respective tank is empty.

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for new and spent fuel assemblies.

Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.4.1 NEW FUEL STORAGE

a. New fuel will normally be stored in the new fuel storage vault or spent fuel pools. The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21-1/8 inches in both directions for the new fuel storage vault and the Spent Fuel Pool "A". The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 13-5/8 inches in both directions for the Spent Fuel Pool "B". This spacing is sufficient to maintain a K effective of less than .95 based on fuel assemblies with an enrichment of 4.3 weight percent U^{235}. When fuel is being stored in the new fuel storage vault, twelve (12) storage locations (aligned in two rows of six locations each; transverse row numbers four and eight) must be left vacant of fissile or moderating material to provide sufficient neutron leakage to satisfy the NRC maximum allowable reactivity value under the optimum low moderator density condition. When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to ensure meeting the NRC maximum allowable reactivity value under the postulated accident condition of a misplaced fuel assembly.

b. New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in an 8 x 8 array storage rack having a nominal center to center distance of 21-1/8 inches. When fuel is being moved in or over the fuel transfer canal, a boron concentration of at least 600 ppmb must be maintained to ensure that, under the postulated accident condition of a misplaced fuel assembly, the maximum reactivity will be less than the NRC maximum allowable reactivity. This applies only when fuel is being stored in the canal.

c. New fuel may also be stored in shipping containers.

5.4.2 SPENT FUEL STORAGE

a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.

b. Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.

c. Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

d. The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:

	South End of Fuel Transfer Canal RB	Spent Fuel Pool A North End of Fuel Handling Building	Spent Fuel Pool B South End of Fuel Handling Building	Dry New Fuel Storage Area Fuel Handling Building
Fuel Assys	64 *	256 **	496 ***	66****
Cores	0.36	1.45	2.8	0.37

NOTES:
* Includes one space for accommodating a failed fuel detection container.
** Includes three spaces for accommodating failed fuel containers.
*** Spent Fuel Pool B contains spent fuel storage racks with a reduced center-to-center spacing of 13 5/8 inches to increase the storage capacity of the pool.
**** Includes twelve spaces which are required to be vacant of fissile or moderating material so that there is sufficient neutron leakage.

e. All of the fuel assembly storage racks provided are designed to Seismic Class 1 criteria to the accelerations indicated below:

	Fuel Transfer Canal in Reactor Building	Fuel Handling Building Dry New Fuel Storage Area And Spent Fuel Pool A	Fuel Handling Building Spent Fuel Pool B
Horiz.	0.76 g	0.38 g	*
Vertical	0.51 g	0.25 g	*

* The "B" pool fuel storage racks are designed using the floor response spectra of the Fuel Handling Building.

f. Fuel in the storage pool shall have a U-235 loading equal to or less than 57.8 grams of U-235 per axial centimeter of fuel assembly.

REFERENCES

(1) FSAR, Section 9.7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 138 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO.: 50-289

1.0 INTRODUCTION

By letter of January 12, 1988 (Ref. 1) GPU Nuclear Corporation (GPU) requested a revision to the Technical Specifications (TS) for Three Mile Island Unit 1 (TMI-1). The TS changes relate to a proposed increase in the U-235 enrichment of fuel which may be placed in various storage racks at TMI for use in TMI-1 operations. There are no proposed changes to the racks.

The TS changes revise the enrichment limit of relevant specifications and also the requirements for the density of boron in the pool water in some of the affected storage areas. The proposal presents criticality analyses for the storage racks to demonstrate compliance with NRC requirements. These analyses were performed for GPU by Stanley Turner (Holtec International) who has been the analyst for a number of fuel storage revision criticality analyses previously reviewed and approved by the staff.

There are four storage areas involved in the enrichment increase. They are (1) the new fuel (NF) storage racks, (2) the Fuel Transfer Canal (FTC) racks, (3) Spent Fuel Pool "A", and (4) Spent Fuel Pool "B". The new fuel area is dry and the other three are flooded. These storage areas are currently licensed for 3.5 percent U-235 enrichment. The request is to change the limit to 4.3 percent. The analysis for the four areas has been done for this enrichment to demonstrate that NRC criteria are met when certain restrictions on the storage are met. The restrictions involve the requirement for at least 600 ppm of boron in the water of the "A" and Fuel Transfer Canal pools, and some required vacant storage spaces in the new fuel racks.

2.0 EVALUATION

2.1 Storage Racks and Fuel

Three of the four storage areas, "A", FTC and NF, have the same rack (square) storage cell lattice configuration and spacing, a 21.125 inch center to center array, and have similar reactivity status (k-effective) for flooded conditions. The largest of these arrays is the "A" rack. Criticality calculations for the "A" system, with minimum neutron leakage, conservatively apply to the FTC and

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NF racks for similar pool water density conditions. However, since the "A" and FTC racks are in pool areas, water density near 1.00 conditions are applicable for meeting NRC criticality criteria, while for the normally dry NF racks it is also necessary to examine low density water moderation conditions.

The "B" racks have a smaller center to center array spacing of 13.625 inches, and each fuel assembly is within a 0.187 inch thick stainless steel box which supplies some neutron absorption. The smaller spacing is designed to achieve a higher storage density and results in the "B" racks providing the limiting value for enrichment, 4.3 percent. The analyses indicate that the other racks could contain higher enrichments, but the present analysis is limited to 4.3 percent.

The fuel to be stored, and that analyzed for this submittal is the standard Babcock & Wilcox 15x15 fuel assembly (with no burnable poison). Minor variations in design may be accommodated via a few sensitivity studies included in the analysis.

2.2 Criticality Analysis Criteria

The current TMI-1 fuel storage system enrichment limits were based on a criterion requiring the k-effective result of criticality analyses to be less than 0.90 with no explicit consideration of uncertainties in the calculation. The current TMI-1 TS 5.4.1 requires a limit of no more than 0.90. Currently, and for over a decade (see NUREG-75/087, 1975, or later editions, NUREG-0800, NRC Standard Review Plan, Sections 9.1.1 and 9.1.2) the NRC storage criterion for spent fuel pools (in flooded conditions) is 0.95, and for new fuel racks (i.e., dry storage) is 0.95 flooded and 0.98 with optimum low density moderation (e.g., from fire fighting sprays). These limits are interpreted as including all relevant uncertainties at a 95/95 (probability/confidence) level. No credit is given for boron in the pool in meeting these limits. Credit may be taken for boron in the pool for analyses of accident conditions (e.g., dropped fuel assembly), however, and such accident events are not considered in combination with optimum moderator conditions in new fuel (dry) storage. The new TMI-1 analyses for the proposed enrichment increase now follow these NRC criteria and positions and are acceptable.

2.3 Analytical Methods

The criticality analyses for the storage areas have used four methodologies (neutronics and cross section packages):

(1) AMPEX-KENO, a Monte Carlo code using the 123 group GAM-THERMOS cross section library and the Nordheim treatment in NITAWL for U-238 resonances.

(2) AMPEX-KENO, using the 27 group SCALE cross sections and the Nordheim treatment.

(3) CASMO-2E, a multigroup transport theory method.

(4) NULIF-PDQ-7, a diffusion theory method with the cross section generation code NULIF.

Method (1) was used only for the low moderator density calculations for the NF racks.

Method (2) was used primarily as a check of CASMO calculations and indicated that CASMO gives conservative values for k-effective for the "A" and "B" pool configurations.

Method (3) was the primary calculation methodology used in the analyses of flooded racks.

Method (4) was used as a third independent methodology and was used to investigate some accident conditions.

All of these methods and associated cross section sets are industry standards, commonly used in criticality calculations. The selection of methods for the various TMI-1 calculations is appropriate. The submittal presents the background material and the benchmarking process and results for these calculations. From these results values for calculational biases and uncertainty have been determined. The benchmarking has investigated the appropriate critical experiments normally investigated for each methodology and application. The results are within ranges to be expected. The review concludes that the methods used for the TMI-1 calculations are thus satisfactory and acceptable.

2.4 Uncertainty and Base Calculation Results

In addition to the calculation method uncertainties, the analyses have investigated the uncertainties associated with the material and geometrical variations of the fuel and rack systems. These have been translated into reactivity increments and have been statistically combined for each of the rack geometries to provide a 95/95 uncertainty value to be added to the base reactivity status calculations. These investigations have examined appropriate variables of the system components and the resulting values and combinations are reasonable and acceptable.

For the "B" racks the CASMO base result was a k-effective of 0.9436 at a pool temperature of 120° C (saturation temperature at the assembly level pool depth), the peak reactivity as a function of temperature found in a sensitivity study. The uncertainty was 0.0056 for a total of 0.9492, within the criterion. The KENO and PDQ-7 results confirmed the base CASMO results.

The "A" rack results, which also apply to the FTC and flooded NF racks, give a value of 0.936 at 20° C (the maximum temperature for that spacing) for the 4.3 percent fuel, and could thus meet the criterion with a larger enrichment if such an analysis were performed.

The NF survey calculations at various low densities using KENO (127) gave a maximum total k-effective of 0.928 at a density of about 7.5 percent for 4.3 percent fuel. For these calculations, however, as a result of preliminary estimates, two rows of storage locations in the racks were assumed to be empty

in order to get results under 0.95. As a result a corresponding restriction on keeping these spaces empty is a part of the TMI-1 TS submitted for these racks. This specification is acceptable. However, it is also required, as indicated in Standard Review Plan 9.1.1, and has been emphasized in staff reviews in recent years, that the empty storage spaces should be blocked to keep accidental access to these spaces to a minimum.

The calculation results and uncertainty analyses for the four storage systems are reasonable and meet the required criteria. They are acceptable.

2.5 Abnormal and Accident Conditions

The submittal examined the reactivity status of conditions considered to be abnormal or accidents in the "A", "B" and FTC storage areas. This included abnormal pool temperature conditions and misplacement (dropping) of a fuel assembly in an improper location. For the NF racks there are no criticality problems possible for such events unless there is also simultaneously near optimum moderation or full flooding. These combinations are not considered by the staff to be sufficiently creditable to be design requirements.

For the "A", "B" and FTC racks the optimum temperature (moderator density) was examined as part of the base calculations and the maximum selected. For a fuel assembly on top of the racks there is sufficient separation so that there is no significant neutronic interaction or reactivity effect. There is insufficient space to place a "misplaced" assembly adjacent to the "B" racks and thus no problem for these racks. For the "A" and FTC racks calculations were done for an assembly immediately adjacent to the rack and a required amount of boron in the pool water to meet the 0.95 requirement was determined. This provided the boron concentration of 600 ppm for the 4.3 percent enrichment submitted for the proposed TS changes.

This analysis of accident conditions and TS requirements is acceptable.

2.6 Technical Specifications

The primary TS changes for this enrichment increase are to TS 5.4.1, "New Fuel Storage" and 5.4.2, "Spent Fuel Storage" (pages 5-6 and 7). The changes involve the replacement of the former reactivity criterion and enrichment limit by 0.95 k-effective and 4.3 percent U-235 respectively. The TS also includes statements on the storage cell separation distances, the geometrical configuration and required vacant spaces of the NF racks, and the requirements for 600 ppm boron in the "A" and FTC pools when fuel is stored or moved. The specification on the limit for the linear weight of U-235 in a stored fuel assembly is also increased to 57.8 gm/cm to match the 4.3 percent enrichment. These changes have been previously discussed in this review and are acceptable.

Also changed is Table 4.1-3 (page 4-10) and the Bases for TS 4.1 (involving pages 4-1 and 2). These introduce the surveillance of the 600 ppm boron concentration into Section 4, "Surveillance Standards". The additions are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

GPU has proposed TS changes for TMI-1 providing for increased U-235 enrichment of fuel in new and spent fuel storage racks. The TS are in accord with staff positions and previous TS approvals. Our review has concluded that appropriate material has been submitted and the TS changes are acceptable. However, it will be necessary to block the vacant storage spaces used in the criticality analysis of the new fuel storage racks.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 25, 1988

Principal Contributor: H. Richings