Distribution
Docket
ORB #3
Local PDR
NRC PDR
VStello
BGrimes
GLear
SSheppard
DVerrelli
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (10)
JMcGough
DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan

APRIL 2 8 1978

Docket No. 50-333

Power Authority of the State
of New York
ATTN: Mr. George T. Berry
General Manager and
Chief Engineer
10 Columbus Circle
New York, New York 10019

Gentlemen:

The Commission has issued the enclosed Amendment No. 35 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application submitted by letter dated April 5, 1978.

This amendment revises the Technical Specifications to increase the operating minimum critical power ratio (MCPR) based on a reanalysis of Cycle 2 operation between End-of-Cycle minus 2000 megawatt days per ton and End-of-Cycle 2. The amendment also adds an additional exposure dependent MCPR at End-of-Cycle 2 minus 1000 MWD/T.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

concurrence subject to paragraph added discussing turbine control valve initial position and scram trip delay time.

Enclosures:
1. Amendment No. 35
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Const. 1

OFFICE➤		ORB #3	ORB #3	OELD	ORB #3	RSB
SURNAME➤		SSheppard	DVerrelli:mjf		GLear	SH for RBaer
DATE➤		4/24/78	4/24/78	4/25/78	4/27/78	4/24/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

cc: Lewis R. Bennett, Assistant General
Manager/General Counsel
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. Peter W. Lyon
Manager-Nuclear Operations
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. J. D. Leonard, Jr.
Resident Manager
James A. Fitzpatrick Nuclear
Power Plant
P. O. Box 41
Lycoming, New York 13093

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

George M. Wilverding, Licensing
Licensing Supervisor
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. Robert P. Jones, Supervisor
Town of Scriba
R. D. #4
Oswego, New York 13126

Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection
Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection
Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated April 5, 1978, 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.

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-59 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 35, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 24, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A Technical Specifications as follows:

Remove pages	Insert pages
29	29
30	30
35	35
103	103

Changes on the revised pages are shown by marginal lines.

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575° F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° F for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% X 1,250 = 1,375 psig), and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% X 1,150 = 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The analysis in NEDO-21619-1 Section 6.3.4 shows that the main steam isolation valve transient, when direct scram is ignored, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is at least 105 psig above the peak pressure produced by the event above. Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected (1,375 psig) overpressure transients. Figure 6-12.3 of NEDO-21619-1 presents the curve produced by this analyses. Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 100 msec.

B. Minimum Critical Power Ratio (MCPR)

During reactor power operation at rated power and flow, the MCPR operating limits shall not be less than those shown below:

FUEL TYPE	MCPR OPERATING LIMIT FOR INCREMENTAL CYCLE 2 CORE AVERAGE EXPOSURE		
	BOC2 to 2GWd/t before EOC2	EOC2-2GWd/t to EOC2-1GWd/t	EOC2-1GWd/t to EOC2
7 x 7	1.22	1.27	1.30
8 x 8	1.20	1.35	1.38

If at anytime during reactor power operation it is determined that the limiting value for MCPR is being exceeded action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type of frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

B. Daily, during reactor power operation, while in the RUN MODE, the peak heat flux and peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specifications 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds the design value of 2.60 for 7 x 7 and 2.42 for 8 x 8 fuel.

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative ($500<P<850$ psig) to the normal EHC oil pressure of 1,600 psig so that, based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the startup and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis for cycle 2 (NEDO 21619-1) for various core exposures are given in Specification 3.1.B.

The ECCS performance analysis assumed reactor operation will be limited to MCPR of 1.18. However, the Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as given in Specification 3.1.B.

resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (NEDO-21619-1 Figures 6.7-1 through 6.7-4) with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than 1.06.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on JAFNPP.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients, 290 msec are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 msec estimated from the scram test results. Approximately 90 msec of each of these intervals result from the sensor and the circuit delay, at this point, the pilot scram valve solenoid de-energizer. Approximately 120 msec later, control rod motion is estimated to actually begin. However, 200 msec is conservatively assumed for this time interval in the transient analysis and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to de-energize the pilot valve scram solenoid is measured during the calibration tests required by Specification 4.1.

The scram times generated at each refueling outage and during operation when compared to scram times generated during pre-operational tests demonstrate that the control rod drive scram function has not deteriorated. In addition, each instant when control rods are scram timed during operation or reactor trips, individual evaluations shall be performed to insure that control rod scram times have not deteriorated.

D. Reactivity Anomalies

During each fuel cycle, excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated April 5, 1978[1] the Power Authority of the State of New York (licensee) requested changes to the Technical Specifications appended to Facility Operating License DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The proposed amendment would increase the operating limit minimum critical power ratio (MCPR) based on a reanalysis of Cycle 2 operation between End-of-Cycle minus 2000 megawatt days per ton (MWD/T) and End-of-Cycle 2. This reanalysis considers more limiting calculational assumptions, reduced safety/relief valve capacity and actual End-of-Cycle 1 exposure data. In addition, the proposed amendment would add an additional exposure dependent MCPR at End-of-Cycle 2 minus 1000 MWD/T.

2.0 DISCUSSION

Based on previously submitted and approved analyses, FitzPatrick has exposure dependent minimum critical power ratio operating limits for two exposure intervals. The exposure intervals are (1) from beginning of cycle (BOC) to 2000 MWD/T before End-of-Cycle (EOC) and (2) from 2000 MWD/T before EOC2 to EOC2. These limits were based on the limiting abnormal operational transient. For that analyses, the turbine trip w/o bypass was limiting. (Ref 1).

The licensee's application of April 5, 1978 included the results of a detailed Cycle 2 reload reanalysis of vessel pressurization and thermal transients with respect to safety margins. The reanalysis of the generator load rejection with no steam by-pass to the condenser and actual end-of-cycle 1 exposure data was -pass,rmed. Other assumptions for this transient were found to affect its outcome: (1) turbine/generator overspeed effects on core flow, (2) turbine control valve initial position and closure time and (3) scram trip delay time. As the result of the reanalysis, the most limiting abnormal transient for cycle 2 was found to be a generator load rejection without steam bypass when the recirculation pumps are assumed to be powered by the main generator. For this limiting transient, there is an accompanying increase in operating limit MCPR over those currently approved.

3.0 EVALUATION

3.1 Fuel Cladding Integrity Safety Limit

The proposed amendment will not change the Safety Limit MCPR (SLMCPR) of 1.06. The SLMCPR is the minimum that is considered necessary to maintain fuel cladding integrity. The SLMCPR is established from a statistical analysis of reactor system and calculational uncertainties. At the SLMCPR, 99.9% of the fuel rods in the core are expected to avoid boiling transition. The calculational method for this evaluation is documented in reference 2 and the SLMCPR of 1.06 for FitzPatrick has been previously reviewed and approved[3].

3.2 Operating Limit MCPR (OLMCPR)

Various transient events will reduce CPR. To assure that the fuel cladding safety limit MCPR of 1.06 is not violated during abnormal operating transients, the most limiting transients have been reanalyzed to determine which transient results in the largest reduction in critical power ratio (i.e., ΔCPR). The licensee's submittal included the results of those transients which produce a significant decrease in MCPR. The types of abnormal operational transients evaluated were reactor pressure increase, feedwater temperature decrease and coolant flow increase.

The most limiting transient from rated conditions in these categories was the load rejection w/o bypass. To assure that this analysis provides adequate safety margin, the licensee's submittal included conservative assumptions to account for turbine generator overspeed effects, turbine control valve fast closure time and scram trip delay time.

If a load rejection were to occur, a turbine/generator overspeed would result. Since the analysis assumed that both recirculation loop pumps are electrically coupled to the main generator, both pumps would also experience the overspeed effect. The increased frequency from the main generator, carried through the motor-generator (M-G) set to the recirculation pumps, would cause increased core flow, with a resulting power increase in the first few seconds of the transient. If the recirculation pumps were automatically switched to an auxiliary or offsite power source, as is the case at FitzPatrick, the pumps would not overspeed and the load rejection transient would not significantly differ from the turbine trip. Thus, the staff finds the assumption on overspeed effects to be conservative.

To a lesser degree than Turbine generator overspeed, turbine control valve (TCV) closure time and scram trip delay time have some effect on the results of the load rejection w/o bypass event. The time for full closure of the turbine control valves has been determined from the system response. FitzPatrick has an Electro-Hydraulic Control (EHC) partial arc TCV system and based on the response time of this system, a 150 msec valve was assumed. This closure time is identical to the time assumed for the FSAR analysis of Genertor Load Rejection w/o bypass on FitzPatrick. The assumption of TCV position, i.e., all valves but one are fully open and that one at 70% full open, maximizes the transient reactivity response, in that it causes the most rapid pressurization. This is due to the combination of effects on scram time and steam flow rate capacity. The position of the valves optimizes the combination of these effects and maximizes the core power response. The valve of 30 msec for the scram delay time is nominally 50 percent longer than that assumed for the turbine trip event. Thus the staff finds the assumption on TCV closure time and scram delay time to be conservative.

These analyses were performed at burnups near and at EOC-2 since the nuclear parameters tend to become more limiting towards EOC for the pressurization and thermal transients. The transient input parameters used for the current Technical Specifications were extrapolated values for EOC-1 conditions; whereas, the reanalysis has included actual EOC-1 exposure data.

The maximum ΔCPRs for the 7x7 and 8x8 fuel which resulted from the transient reanalyses are summarized below:

	EOC 2 minus 2000 MWD/T to EOC 2 minus 1000 MWD/T		EOC 2 minus 1000 MWD/T to EOC 2	
	7x7 fuel	8x8 fuel	7x7 fuel	8x8 fuel
Load Rejection w/o Bypass	.21	.29	.24	.32

Addition of these ΔCPRs to the SLMCPR of 1.06 gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit should this limiting transient occur.

The SLMCPR is established from a statistical propagation of uncertainties which results in a probabilistic criterion on boiling transition,[2] i.e., 99.9% of rods in the core avoid boiling transition. The method of analysis in reference 2 conservatively established the uncertainties for the entire reactor cycle, and therefore, is applicable to the period of time covered by the proposed change. The change in exposure intervals does not affect the statistical method or uncertainties and the SLMCPR is not changed.

The change to an intermediate exposure interval does affect the ΔCPR evaluation. The ΔCPR is the change in CPR which results from an abnormal operating transient. The transient analysis and ΔCPR evaluation vary with exposure. Therefore an evaluation for the proposed exposure interval has been properly submitted[1].

Transient analyses results vary with exposure. Such exposure-dependent variation is due to the change in transient input paramaters (e.g., dynamic void coefficient, Doppler coefficient, scram worth, and scram reactivity). The scram reactivity curves are the major cause of the variation with exposure. Towards EOC, the control rods tend to be positioned out of the core. Thus, the scram reactivity insertion rate toward the end of cycle is slower because the control rods do not get into the core as rapidly as earlier in the cycle. Therefore, with increasing cycle burnup the ΔCPR necessary to accommodate various transients is increased. The resulting changes in ΔCPR and OLMCPR have been presented in the licensee's submittal.

We have reviewed the licensee's submittal. In this review we found that the licensee has used the same or more conservative methods for the transient analysis and the ΔCPR evaluation as previously reviewed and approved for FitzPatrick. The use of exposure-dependent OLMCPR has also been reviewed and approved for FitzPatrick. On this basis we find the proposed change in Technical Specification acceptable.

The licensee's submittal included a reanalysis of the operating transient which causes the most severe reactor isolation. The licensee committed to perform this analysis based on his discovery that the inlet piping to the safety/relief valves was slightly smaller than the inside diameter of the inlet section of the Target Rock valves (Ref 4). The licensee calculated that the effect of this discrepency was to produce a 2.3% increase in pipeline pressure drop at 1100 psia which reduces the valve capacity (lbm/sec) by the same amount. Calculations of the limiting pressurization transient, as MISV closure with indirect flux scram, indicate that the pressure margin between the peak vessel pressure and the ASME code limit of 1375 psig is 118 psi. The 118 psi margin is within the margin set forth in the bases of the current Technical Specifications related to reactor coolant system overpressure conditions and satisfies the American Society of Mechanical Engineers Vessel Pressure Code specifications.

3.3 ECCS Appendix K Analysis

The licensee assessed the effect of the 2.3% reduction in relief valve capacity on a loss of coolant accident (LOCA) analysis. The limiting large-break LOCA for FitzPatrick, which produces the highest peak clad temperature (PCT) is a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break (Ref 3). For this accident, the reduced relief valve capacity has negligible effects since the reactor vessel depressurizes very rapidly. For the limiting small line break (0.07 ft^2 in the recirculation pump suction line) the licensee calculated, based on a generic study, that the effect of a 2.3% reduction in relief capacity would increase the previously calculated PCT by 25°F. Thus the maximum PCT during a small break accident would be 1285°F.

Based on our previous safety evaluation of LOCA Analysis (Ref 3) and the licensee's submittal, we conclude that no revision to the currently approved maximum average planar heat generation limits is required.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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 28, 1978

References

1. Letter, Power Authority of the State of New York (Berry) to USNRC (Lear) transmitting NEDO-21619-1, April 5, 1978.

2. General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application, General Electric Company BWR Systems Department, November 1973 (NEDE-10958, Class III).

3. Safety Evaluation by NRR Supporting Amendment No. 30 to Facility Operating License No. DPR-59, September 16, 1977.

4. Prompt Reportable Occurrence Letter, PASNY (Pasternak) to USNRC (Grier), January 5, 1978.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 35 to Facility Operating License No. DPR-59, issued to Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant (the facility) located in Oswego County, New York. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specifications to increase the operating minimum critical power ratio (MCPR) based on a reanalysis of Cycle 2 operation between End-of-Cycle minus 2000 megawatt days per ton and End-of-Cycle 2. The amendment also adds an additional exposure dependent MCPR at End-of-Cycle 2 minus 1000 MWD/T.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated April 5, 1978, (2) Amendment No. 35 to License No. DPR-59, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oswego County Office Building, 46 East Bridge Street, Oswego, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 28th day of April 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors