September 15, 1977

Docket No.: 50-333

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Power Authority of the State
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ATTN: Mr. George T. Berry
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Gentlemen:

The Commission has issued the enclosed Amendment No. 28 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 for amendment submitted by letter dated June 17, 1976, and staff discussions.

The amendment provides for changes in (1) primary containment atmospheric monitor surveillance requirements; (2) radiation level setpoints for various building ventilation systems; and (3) the reactor coolant leakage monitoring system in the drywell.

Copies of the Safety Evaluation/Environmental Impact Appraisal and the Notice of Issuance/Negative Declaration are enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 28
2. Safety Evaluation/Environmental Impact Appraisal
3. Notice/Negative Declaration

cc w/enclosures: See next page

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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. 28
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) sworn to June 15, 1976, 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. 28, 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

Gerald B. Zwetzig for
Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 15, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Remove pages 59, 60, 74, 142, 151, 152, 181, 209, and 210 of the Appendix A Technical Specifications and insert the attached pages 59, 60, 74, 142, 142a, 151, 152, 162a, 181, 209, and 210. Changes on these pages are shown by marginal lines. Pages 60, 152, and 209 are unchanged and are included for convenience only.

the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e, only one instrument channel out of service.

Two air ejector offgas monitors are provided and when their trip point is reached, cause an isolation of the air ejector offgas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a 15 min. delay before the air ejector offgas isolation valve is closed. This delay is accounted for by the 30 min. holdup time of the offgas before it is released to the stack. Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the instantaneous stack release rate limit given in Environmental Technical Specification 2.3.B is not exceeded.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the Standby Gas Treatment System. The monitors are located as follows: two in the reactor building ventilation exhaust duct and two in refuel floor ventilation exhaust duct. Each pair is considered a separate system. The trip logic consists of any upscale trip on a single monitor or a downscale trip on both monitors in a pair to cause the desired action.

Trip settings of 2.7×10^5 cpm for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and Standby Gas Treatment System operation so that all of the activity released during the refueling accident is processed by the Standby Gas Treatment System.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The alarm unit in each integrator is set to annunciate before the values specified in Specification 3.6.D are exceeded.

For each parameter monitored, as listed in Table 3.2-6, there are two channels of instrumentation. By comparing readings between the two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

TABLE 3.2-4

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action (2)
1	Refuel Area Exhaust Monitor	$\leq 2.7 \times 10^5$ cpm (5)	2 Inst. Channels	A or B
1	Reactor Building Area Exhaust Monitors	$\leq 2.7 \times 10^5$ cpm (5)	2 Inst. Channels	B
1	Off-gas Radiation Monitors	$\leq 7 \times 10^4$ mR/hr (3)	2 Inst. Channels	C
1	Turbine Bldg. Exhaust Monitors	$\leq 1.8 \times 10^5$ cpm (5)	2 Inst. Channels	C
1	Radwaste Bldg. Exhaust Monitor	$\leq 6.7 \times 10^5$ cpm (5)	2 Inst. Channels	C
1	Main Control Room Ventilation Monitor	$\leq 4 \times 10^3$ cpm (6)	1 Inst. Channels	D
2	Mechanical Vacuum Pump Isolation	≤3 times normal full power background	4 Inst. Channels	E
1	Liquid Radwaste Discharge Monitor	(4)	1 Inst. Channel	F

NOTES FOR TABLE 3.2-4

1. Whenever the systems are required to be operable, there shall be two operable or tripped instrument channels per trip system. From and after the time it is found that this cannot be met, the indicated action shall be taken.

2. Action

   A. Cease operation of the refueling equipment.
   B. Isolate secondary containment and start the Standby Gas Treatment System.
   C. Refer to Section 2.3.B.4 of Environmental Technical Specification.
   D. Control Room isolation is manually initiated.
   E. Uses same sensors as Primary Containment Isolation on high main steam line radiation. Table 3.2-1.
   F. Refer to Environmental Technical Specification 2.3.A.3.

3. Refer to Specification 2.3.B of the Environmental Technical Specifications.

4. Trip setting to correspond to Specification 2.3.A of the Environmental Technical Specifications.

5. Conversion factor is 9.0×10^7 cpm = 1 µci/cc

6. Conversion factor is 8.15×10^7 cpm = 1 µci/cc

made or found to be inoperable for any reason, continued reactor operation is permissible for the succeeding seven days, unless such subsystem is sooner made operable.

b. From and after the time that a redundant component of either subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible for the succeeding 30 days, unless the component is sooner made operable.

3. The Drywell Continuous Atmosphere Radioactivity Monitoring System shall be operable during reactor power operation.

a. The monitoring system shall be considered operable if at least one monitor is operable.

b. From and after the time the Drywell Continuous Atmosphere Radioactivity Monitoring System is made or found to be inoperable for any reason, continued reactor power operation is permissible provided:

1. The requirements of 3.6.D.2 are satisfied and

4.6 (cont'd)

3. Drywell Continuous Atmosphere Radioactivity Monitoring System instrumentation shall be functionally tested and calibrated as specified in Table 4.6-2.

2. An appropriate grab sample is obtained and analyzed at least once per 96 hours.

   In the event of a significant change in Reactor Coolant Leakage the appropriate grab sample will be obtained and analyzed at least once per 24 hours.

4. If Specification 3.6.D cannot be met, the reactor shall be placed in a cold condition within 24 hr.

E. Safety and Safety/Relief Valves

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, the safety mode of all safety/relief valves shall be operable, except as specified by Specification 3.6.E.2. The Automatic Depressurization System valves shall be operable as required by Specification 3.5.D.

E. Safety and Safety/Relief Valves

1. At least one half of all safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. The safety/relief valve settings shall be set as required in Specification 2.2.B. All valves shall be tested every two operating cycles.

leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.D on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.6.D, the experimental and analytical data suggest a reasonable margin of safety such that leakage of this magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the Plant should be shut down to allow further investigation and corrective action.

The capacity of the drywell sump pumps is 100 gpm, and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of the Reactor Coolant Leakage Detection System will be evaluated during the first 5 yr of plant operation, and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detectors are capable of detecting a leak on the order of 3,500 lb/hr. The system performance will be evaluated during the first 5 yr of plant operation, and the conclusions of the evaluation will be reported to the NRC.

The reactor coolant leakage detection systems consist of the drywell sump monitoring system and the drywell continuous atmosphere monitoring system. The drywell continuous atmosphere monitoring system utilizes a three-channel monitor to provide information on particulate, iodine and noble gas activities in the drywell atmosphere. Two independent and redundant systems are provided to perform this function. This system supplements the drywell sump monitoring system in detecting abnormal leakage that could occur from the reactor coolant system. In the event that the drywell continuous atmosphere monitoring system is inoperable, grab sample will be taken on a periodic basis to monitor drywell activity.

E. Safety and Relief/Safety Valves

Experiences in safety valve operation show that the testing of 50 percent of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1 percent of design pressure. An analysis has been performed which shows that with all safety valves set 1 percent higher, the reactor coolant pressure safety limit of 1,375 psig is not exceeded.

The relief/safety valves have two functions; i.e., power relief or self-actuated by high pressure. Power relief is a solenoid actuated function (Automatic Depressurization System) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to open. This function is discussed in Specification B.3.5.D. In addition, the valves can be operated manually.

The safety function is performed by the same relief/safety valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III - Nuclear Vessels, requires that these bellows be monitored for failure, since this would defeat the safety function of the relief/safety valve.

It is realized that there is no way to repair or replace the bellows during operation, and the plant must be shut down to do this. The 30-day and 7-day periods to do this allow the operator flexibility to choose his time for shutdown; meanwhile, because of the redundancy present in the design and the continuing monitoring of the integrity of the other valves, the overpressure pressure protection has not been compromised in either case. The auto-relief function would not be impaired by a failure of the bellows. However, the self-actuated overpressure safety function would be impaired by such a failure. There is no provision for testing the bellows leakage pressure switch during plant operation. The bellows leakage pressure switches will be removed and bench checked once/operating cycle. These bench checks provide adequate assurance of bellows integrity.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the relief/safety and safety valves are

Table 4.6-2

Minimum Test and Calibration Frequency for Drywell Continuous Atmosphere Radioactivity Monitoring System

Inst. Channel	Inst. Functional Test	Calibration	Sensor Check
1. Air Particle Analyzer	None	Once/3 mos.	once/day
2. Gaseous Activity Analyzer	None	Once/3 mos.	once/day
3. Iodine Analyzer	None	Once/3 mos.	once/day

8. Primary Containment Atmosphere Monitoring Instruments

a. Primary containment atmosphere shall be continuously monitored for hydrogen and oxygen when the containment integrity is required.

B. Standby Gas Treatment System

1 Except as specified in 3.7.B.2 below, both circuits of the Standby Gas Treatment System shall be operable at all times when secondary containment integrity is required.

8. Primary Containment Atmosphere Monitoring Instruments

a. Instrumentation shall be functionally tested and calibrated as specified in Table 4.7-1.

B. Standby Gas Treatment System

1. Standby Gas Treatment System surveillance shall be performed as indicated below:

a. At least once per operating cycle, it shall be demonstrated that:

(1.) Pressure drop across the combined high-efficiency and charcoal filters is less than 5.7 in. of water at 6,000 scfm and

(2.) 39KW heater outlet shall not have greater than 70 % relative humidity at 6000 scfm.

NOTES FOR TABLE 3.7-1 (CONT'D)

9. Coincident low reactor water level signal "G" and low reactor pressure signal "T" open LPCI valves, except that recirculation line break signal "H" overrides to close LPCI valves on borken side and automatically opens the LPCI valves in the opposite loop. Special interlocks permit testing these valves with manual switch during any mode of reactor operation except when coincident signals "G" and "T" are present.

10. Coincident signals "G" and "T" open valves. Special interlocks permit testing these valves by manual switch except when automatic signals are present.

11. Normal status position of valve (open or closed) is the position during normal power operation of the reactor (see "Normal Status" column).

12. The specified closure rates are as required for containment isolation only.

13. Minimum closing time is based on valve and line size. 209

14. Signal "A" or "F" causes automatic withdrawal of TIP probe. When probe is withdrawn, the valve automatically closes by mechanical action.

15. Reactor building ventilation exhaust high radiation signal "Z" is generated by two trip units. This required one unit at high trip or both units at down scale (instrument failure) trip, in order to initiate isolation.

TABLE 4.7-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTAINMENT MONITORING SYSTEMS

Instrument Channel	Instrument Functional Tests	Calibration	Sensor Check
1. Hydrogen Analyzer	None	Once/3 months	Once/day
2. Oxygen Analyzer	None	Once/3 months	Once/day

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 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

Introduction

By an application for amendment submitted by letter dated June 17, 1976, the Power Authority of the State of New York (the licensee), proposed changes to the Appendix A Technical Specifications appended to Facility Operating License No. DPR-59, for the James A. FitzPatrick Nuclear Power Plant (the facility). Changes were requested in three areas: (1) primary containment atmospheric monitor surveillance requirements; (2) drywell leakage detection equipment surveillance requirements; and (3) airborne radiation monitor setpoints in the ventilation exhaust ducts in the refuel area, reactor building, turbine building, and radwaste building.

During the course of our review of these proposed changes, we had several discussions with licensee representatives to clarify issues and to correct minor errors.

I. Safety Evaluation

The three requested Technical Specification changes noted above will be discussed separately.

Primary Containment Atmospheric Monitor

The current Technical Specifications require containment atmosphere to be continuously monitored when containment integrity is required. The proposed amendment would change the requirement to define that hydrogen and oxygen concentrations in containment would be continuously monitored when containment integrity is required. Since this is a clarification of the current surveillance requirements, we find the change acceptable.

Leakage Detection System

Reactor coolant leakage detection indications are currently derived from a drywell air particulate monitor and the drywell sump monitor. There is also a moisture analyzer which is proposed to be eliminated since it continuously indicates 100% relative humidity. It cannot distinguish any change in coolant leakage rate.

There are currently two redundant, three-channel monitors that provide information on particulate, iodine, and noble gas activity. The Technical Specifications require surveillance on only particulate. There is no surveillance presently being conducted on iodine and noble gas activity information received by the monitors.

Regulatory Guide 1.45 recommends at least three separate primary coolant leak detection methods. Currently these methods are: (1) sump level and sump flow monitoring; (2) particulate monitoring; and (3) moisture analyzing. Under the proposed change, "moisture analyzing" would be eliminated and in its place monitoring of gaseous radioactivity would be performed through surveillance of iodine and noble gas activity. This is an improved method of monitoring reactor coolant leakage as compared to using the moisture analyzer. Thus, the proposed change would provide greater assurance of detecting possible leakage in the containment and therefore provide for an increase in safety. We, therefore, find the change acceptable.

Airborne Radiation Monitor Setpoints

The FitzPatrick Technical Specifications incorporate airborne radiation monitor setpoints for the Refuel Area exhaust, the Reactor Building exhaust, the Turbine Building exhaust, and the Radwaste Building exhaust. Each exhaust separately exhausts gases from the respective areas to the atmosphere. The present setpoints are 900 counts per minute (CPM) which causes each applicable exhaust to be closed automatically.

For the Reactor Building and the Refueling Area, a trip at the setpoint causes the Standby Gas Treatment System to operate; and for the Turbine Building and Radwaste Building, a trip at the setpoint causes a shutdown of the respective building's ventilation system. In addition, the facility's Emergency Plan requires the evacuation of all station personnel to one central assembly area in the event any exhaust monitor initiates a trip at the setpoint.

The licensee states that the setpoints are unnecessarily low and that he has experienced many inadvertent isolations of the pertinent areas and evacuation.

The licensee proposes to increase the setpoints of the ventilation monitors to the following levels:

	CPM	μCi/SEC
Refueling area	$2.7X10^5$	$9.91X10^4$
Reactor Bldg.	$2.7X10^5$	$9.91X10^4$
Turbine Bldg.	$1.8X10^5$	$8.92X10^4$
Radwaste Bldg.	$6.7X10^5$	$1.16X10^5$

The change also would implement an early alarm whereby the monitors would alarm at 4% of the above values in which case evacuation of all station personnel to one central area would be initiated. In calculating these setpoints, it was assumed that one exhaust monitor had reached the "isolation" setpoint and the other three were at the "alarm" level. This was considered conservative since any other assumed event would have a remote probability of occurrence. If the radioactive releases were only controlled by the Technical Specifications on the proposed setpoints, the proposed setpoints would allow greater radioactive releases than those allowed by the current setpoints. However, the releases are limited by Section 2.3.B of the Environmental Technical Specifications to a small fraction of those allowed by 10 CFR Part 20. The Environmental Technical Specifications are not proposed to be changed and they provide for limiting the conditions of operation to that which will assure operation within the specified limits. Raising the setpoints of the ventilation exhaust monitors will not allow the licensee to discharge more activity in a year than the maximum allowed nor to discharge concentrations greater than the maximum allowed. Consequently, there will be no appreciable effect on the health and safety of the public from this action. We conclude the change in the setpoints is acceptable.

Conclusion on Safety

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.

II. Environmental Impact Appraisal

Potential environmental impacts associated with the proposed Technical Specification changes are those associated with raising the radiation monitor trip setpoints. Although raising the monitor setpoints will allow greater radioactive releases than those allowed by the current setpoints, the releases will be limited by Section 2.3.B of the Environmental Technical Specificationsto a small fraction of those allowed by 10 CFR Part 20. Raising the setpoints of the ventilation exhaust radiation monitors will not allow the licensee to discharge more activity in a year than the maximum allowed nor to discharge concentrations greater than the maximum allowed. Consequently there will be no appreciable effect on the environment or the health and safety of the public from this action. No increase in total effluent activity greater than that evaluated in the FitzPatrick Final Environmental Statement (FES) is expected.

Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for the FitzPatrick Nuclear Power Plant. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Dated: September 15, 1977

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 AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 28 to Facility Operating License No. DPR-59, issued to the 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.

The amendment provides for changes in (1) primary containment atmospheric monitor surveillance requirements; (2) radiation level setpoints for various building ventilation systems; and (3) the reactor coolant leakage monitoring system in the drywell.

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 prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action other than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility.

For further details with respect to this action, see (1) the application for amendment submitted by letter dated June 17, 1976, (2) Amendment No. 28 to License No. DPR-59, and (3) the Commission's related Safety Evaluation/Environmental Impact Appraisal. 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 E. 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 15th day of September 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Gerald B. Zwetzig

Gerald B. Zwetzig, Acting Chief
Operating Reactors Branch #4
Division of Operating Reactors