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Docket No. 50-333

NOV 22 1978

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

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 43 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 reload application submitted by letter dated August 18, 1978.

Your application for amendment requested approval of the following modifications and/or analyses:

1. Addition of a new Average Power Range Monitor (APRM) scram trip logic;

2. Installation of a new Analog Transmitter/Trip Unit System (ATTUS) to the Reactor Protection System;

3. Regrouping and setpoint changes to the Safety Relief Valves; and

4. The analysis to justify the above modifications as well as the analysis to justify the reload itself.

Item (1), the APRM modification, has not been approved. Per verbal communications with your licensing personnel, approval is not necessary for plant startup.

Item (2), the ATTUS modification, has been approved by licensing Amendment No. 42 issued this date.

With respect to item (3), SRV modifications, by letter dated March 20, 1978, we requested a plant unique assessment of multiple-subsequent safety relief valve actuations. By a series of letters, the last two of which were dated September 28, 1978 and November 14, 1978, you provided the analysis utilizing the criteria specified. The results as well as

Cont'd

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the calculational techniques of this Mark I containment short term analysis have been reviewed. The proposed SRV setpoints assure that the analysis of the containment structure satisfies the Mark I Short Term Acceptance criteria. Therefore, we conclude that your analysis is acceptable on an interim basis awaiting final resolution by the Mark I Containment Long Term Program.

Regarding the reload itself (item 4) we have reviewed the General Electric submittal, "Supplemental Reload Licensing Submittal for the James A. FitzPatrick Nuclear Power Plant for Reload No. 2", NEDO-24129, dated June 1978. The justification contained therein as well as the supporting information provided in Attachment B to your August 18, 1978 letter has been found acceptable. The proposed Technical Specification changes necessary for cycle 3 operation are enclosed.

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

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 43 to DPR-59
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

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DATE	11/20/78	11/21/78	11/20/78	11/20/78	11/20/78	11/ /78

Mr. George T. Berry

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Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

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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. 43
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 August 18, 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. 43, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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 change.

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surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ± 25 percent. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

1. Minimum critical power ratio (MCPR)-Ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power as calculated by application of the GEXL correlation (Reference NEDE-10958).

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 18.5 KW/ft for 7x7 bundles and 13.4 KW/ft for 8x8 and 8x8R bundles.

3. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

4. Transition Boiling - Transition boiling means the boiling region between nucleate and film boiling. Transition boiling is the region in which both nucleate and film boiling occur intermittently with neither type being completely stable.

V. Electrically Disarmed Control Rod

To disarm a rod drive electrically, the four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred. Electrical disarming does not eliminate position indication.

W. High Pressure Water Fire Protection System

The High Pressure Water Fire Protection System consists of: a water source and pumps; and distribution system piping with associated post indicator valves (isolation valves). Such valves include the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler or water spray subsystem.

X. Staggered Test Basis

A Staggered Test Basis shall consist of:

a. A test schedule for a systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

1.1 SAFETY LIMITS

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit, hereafter called the Safety Limit.

2.1 LIMITING SAFETY SYSTEM SETTINGS

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. IRM - The IRM flux scram setting shall be set at ≤ 120/125 of full scale.

1.1 (cont'd)

B. <u>Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig)</u>

When the reactor pressure is $\leq$ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. <u>Power Transient</u>

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

A.1.b. <u>APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)</u>

APRM - The APRM flux scram setting shall be $\leq$ 15 percent of rated neutron flux, with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. <u>APRM Flux Scram Trip Settings (Run Mode)</u>

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be:

$$S \leq 0.66\ W + 54\%$$

where:

S = Setting in percent of rated thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 x 10^6 lb/hr)

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

1.1 (cont'd)

D. Reactor Water Level (Hot or Cold Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 18 in. (-146.5 in. indicated level) above the top of the active fuel when it is seated in the core.

2.1 (cont'd)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%) \left[\frac{FRP}{MFLPD}\right]$$

where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 and 8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

(2) Fixed High Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

$$S \leq 120\%\ \text{Power}$$

1.1 (cont'd)

2.1 (cont'd)

A.1.d. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

$$S \leq 0.66\ W + 42\%$$

where:

S = Rod block setting in percent of thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals (34.2×10^6 lb/hr))

In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 42\%) \left[\frac{FRP}{MFLPD}\right]$$

where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 and 8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

1.1 BASES

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding, perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the MCPR operating limits specified for the normal operating conditions in specification 3.1.B, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided at the beginning of each fuel cycle. Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the Safety Limit would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Fitzpatrick operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (Safety Limit) operation is constrained to a maximum LHGR = 18.5 kw/ft for 7x7 fuel and 13.4 kw/ft for 8x8 and 8x8R fuel. At 100% power, this limit is reached with a maximum fraction of limiting power density (MFLPD) equal to 1.0. In the event of operation with a MFLPD greater than the fraction of rated power (FRP), the APRM scram and rod block settings shall be adjusted as required in Specifications 2.1.A.1.c and 2.1.A.1.d.

B. <u>Core Thermal Power Limit (Reactor Pressure $<$ 785 psig)</u>

At pressures below 785 psig the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 0 psig to 785 psig indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

2.1 BASES (cont'd)

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the Safety Limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

b. APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

c. APRM Flux Scram Trip Setting (Run Mode)

The APRM flux scram trip in the run mode consists of a flow referenced scram setpoint and a fixed high neutron flux scram setpoint. The APRM flow referenced neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions. This prevents spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Examples of events which can result in momentary neutron flux spikes are momentary flow changes in the recirculation system flow, and small pressure disturbances during turbine stop valve and turbine control valve testing. These flux spikes represent no hazard to the fuel since they are only of a few seconds duration and less than 120% of rated thermal power.

The APRM flow referenced scram trip setting at full recirculation flow is adjustable up to 117% of

c. <u>APRM Flux Scram Trip Setting (Run Mode) (cont'd)</u>

rated power. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 80°F feedwater heating event, than would result with the 120% fixed high neutron flux scram trip. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower Operating Limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the cycle.

The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow referenced scram.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.c, when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by either (1) reducing the APRM scram and rod block settings or (2) adjusting the indicated APRM signal to reflect the high peaking condition.

Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR will be greater than the Safety Limit when the transient is initiated from the MCPR operating limits provided in Specification 3.1.B.

d. <u>APRM Rod Block Trip Setting</u>

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationships therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

2. <u>Reactor Water Low Level Scram Trip Setting (LLI)</u>

The reactor low water level scram is set at a point which will assure that the water level used in the Bases for the Safety Limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

3. Turbine Stop Valve Closure Scram Trip Settings

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains along the Safety Limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the turbine bypass. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator dise dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar and no more severe than for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in Section 14.5 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first stage pressure.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the Reactor Mode Switch be in the Startup position where protection of the fuel cladding integrity safety limit is provided by the APRM high neutron flux scram and the IRM. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at ≤ 10 percent valve closure, there is no increase in neutron flux.

6. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation minimum limit at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitute an unsafe condition.

C. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor, "NEDO-10802, Feb., 1973.

2. Licensing Topical Reports, "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDO-24011-2, March, 1978.

RATED THERMAL POWER = 2436

RATED CORE FLOW = 77.0×10^6

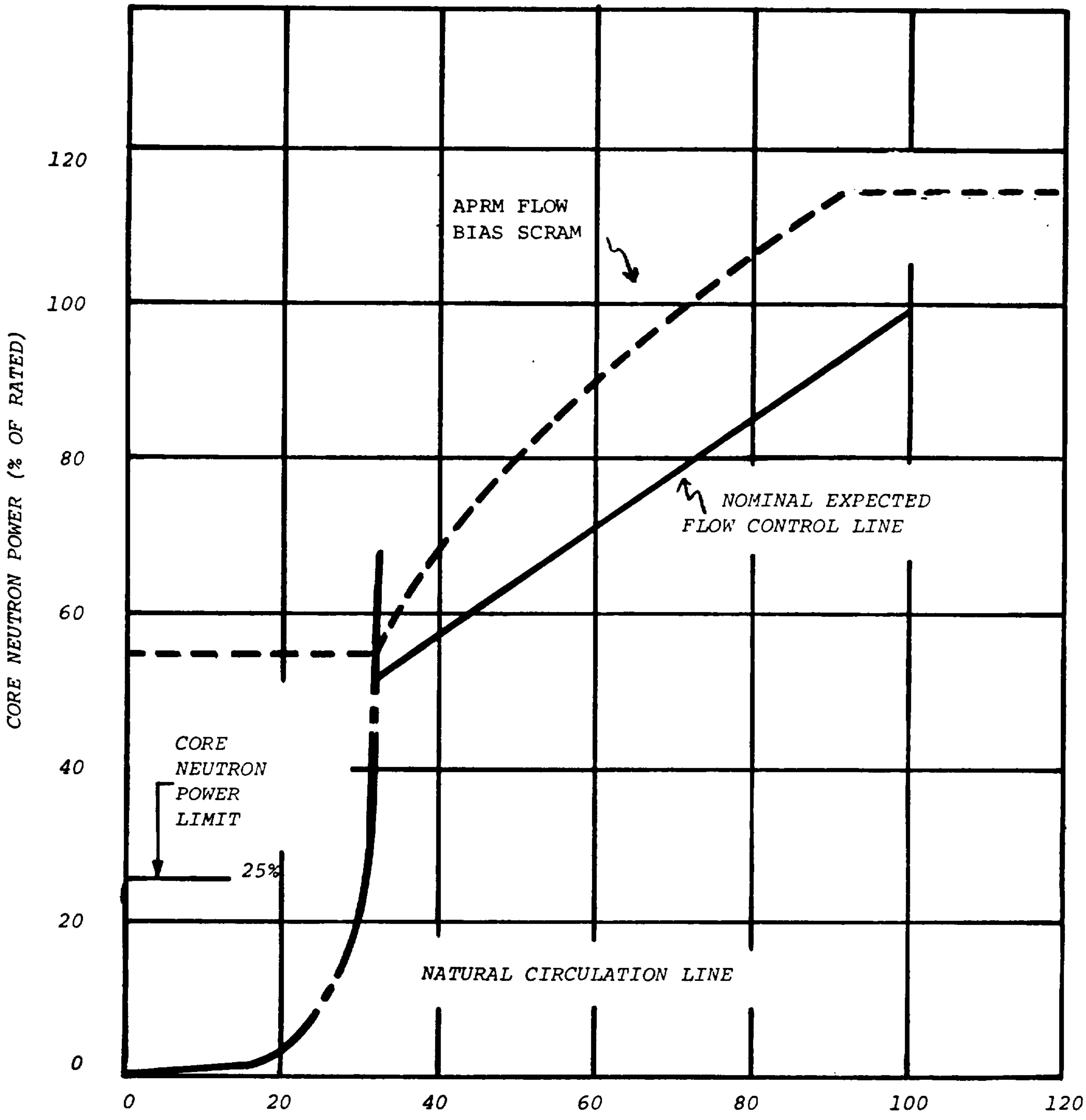

FIGURE 1.1-1 APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

1. The reactor coolant system pressure shall not exceed 1,325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

1. The Limiting Safety System setting shall be specified below:

   A. Reactor coolant high pressure scram shall be $\leq$ 1,045 psig.

   B. Reactor coolant system safety/relief valve nominal settings shall be as follows:

      Safety/Relief Valves

      2 valves at 1090 psig
      2 valves at 1105 psig
      7 valves at 1140 psig

      The allowable setpoint error for each safety/relief valve shall be $\pm$ 1 percent.

1.2 (cont'd)

2. The reactor vessel dome pressure shall not exceed 75 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.

2.2 (cont'd)

2. Action shall be taken to decrease the reactor vessel dome pressure below 75 psig or the shutdown cooling isolation valves shall be closed.

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-24129, "Supplemental Reload Licensing Submittal for the James A. FitzPatrick Nuclear Power Plant for Reload No. 2", June 1978, as amended by NEDO-24129-1, Supplement 1, September 1978, 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 above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. Figure 3 in NEDO-24129-1 presents the curve produced by this analysis. 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 3 CORE AVERAGE EXPOSURE		
	BOC3 to 2GWd/t before EOC3	EOC3-2GWd/t to EOC3-1GWd/t	EOC3-1GWd/t to EOC3
7x7	1.21	1.25	1.30
8x8	1.22	1.33	1.37
8x8R	1.20	1.33	1.37

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. Maximum Fraction of Limiting Power Density (MFLPD)

The MFLPD shall be determined daily during reactor power operation at ≥ 25% rated thermal power and the APRM high flux scram and Rod Block trip settings adjusted if necessary as required by Specifications 2.1.A.1.c and 2.1.A.1.d, respectively.

3.1 (Cont'd)

If anytime during reactor operation greater than 25% of rated power it is determined that the limiting value for MCPR is being exceeded, action shall then be initiated within fifteen (15) minutes to restore operation to within the prescribed limits. If the MCPR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the MCPR is returned to within the prescribed limits. For core flows other than rated, the MCPR operating limit shall be multiplied by the appropriate k_f factor where k_f is as shown in figure 3.1.1.

C. MCPR shall be determined daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

3.1 BASES (cont'd)

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 3 (NEDO-24129 and NEDO-24129-1, Supplement 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.

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4.1 BASES (cont'd)

is meaningful to the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semi-conductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a maximum drift of 0.4 percent could occur, thus providing for adequate margin.

For the APRM System, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every 7 days.

Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1-1 and 4.1-2 indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable.

B. The MFLPD is checked once per day to determine if the APRM scram requires adjustment. Only a small number of control rods are moved daily and thus the MFLPD is not expected to change significantly and thus a daily check of the MFLPD is adequate.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating twice a week using heat balance data and by calibrating individual LPRM's every 1000 effective full power hours, using TIP traverse data.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	≤ 120/125 of full scale	X	X		8 Instrument Channels	A
3	IRM Inoperative		X	X		8 Instrument Channels	A
2	APRM Neutron Flux-Startup (15)	≤ 15% Power	X	X		6 Instrument Channels	A
2	APRM Flow Referenced Neutron Flux (12) (13) (14) (Not to exceed 117%)	$S_n \leq (0.66W+54\%) \times \left[\frac{FRP}{MFLPD}\right]$			X	6 Instrument Channels	A or B
2	APRM Fixed High Neutron Flux (14)	≤ 120% Power			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B

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TABLE 3.1-1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)	Startup	Run		
2	APRM Downscale	≥ 2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	≤ 1045 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤ 2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥ 12.5 in. indicated level	X	X	X	4 Instrument Channels	A
2	High Water Level in Scram Discharge Volume	≤ 36 gal	X(2)	X	X	4 Instrument Channels	A
2	Main Steam Line High Radiation	≤ 3 x normal full power background	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤ 10% valve closure	X(3)(5)	X(3) (5)	X(5)	8 Instrument Channels	A
2	Turbine Control Valve Fast Closure	500 < P < 850 psig Control oil pressure between fast closure solenoid and disc dump valve			X(4)	4 Instrument Channels	A or C

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TABLE 3.1-1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
4	Turbine Stop Valve Closure	≤10 % valve closure			X(4)(5)	8 Instrument Channels	A or C

NOTES OF TABLE 3.1-1

1. There shall be two operable or tripped trip systems for each function, except as specified in 4.1.D. From and after the time that the minimum number of operable instrument channel for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place Mode Switch in the Startup Position within eight hours.

   C. Reduce power to less than 30 percent of rated.

2. Permissible to bypass, in Refuel and Shutdown positions of the Reactor Mode Switch.

3. By passed when reactor pressure is < 1005 psig.

4. Bypassed when turbine first stage pressure is less than 217 psig or less than 30 percent of rated.

5. The design permits closure of any two lines without a scram being initiated.

6. When the reactor is subcritical and the reactor water temperature is less than 212 ^{o}F, only the following trip functions need to be operable:

   A. Mode Switch in Shutdown

   B. Manual Scram

NOTES OF TABLE 3.1-1 (Cont'd)

C. High Flux IRM

D. Scram Discharge Volume High Level

E. APRM 15% Power Trip

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel

9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.

10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.

11. See Section 2.1.A.1.

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP).

where:

FRP = Fraction of rated thermal power (2436 MWt)

MFLPD = Maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 and 8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used

W = Loop Recirculation flow in percent of rated (rated is 34.2×10^6 lb/hr)

S_n = Scram setting in percent of initial

13. The Average Power Range Monitor scram function is varied (Figure 1.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

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TABLE 3.1-1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (Cont'd)

14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is placed in the Run position.

Table 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Maximum frequency once/week
APRM High Flux			
Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source	Every refueling outage
LPRM Signal	B	TIP System Traverse	Every 1000 effective full power hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 months
High Drywell Pressure	A	Standard Pressure Source	Every 3 months
Reactor Low Water Level	A	Pressure Standard	Every 3 months
High Water Level in Scram Discharge Volume	A	Note (5)	Note (5)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 months
Turbine Plant Stage Pressure Permissive	A	Standard Pressure Source	Every 6 months
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	Once/operating cycle

Table 4.1-2 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency (2)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Reactor Pressure Permissive	A	Standard Pressure Source	Every 6 months

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the Bases of this Specification.

2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Response time is not a part of the routine instrument channel test but will be checked once per operating cycle.

5. Actuation of these switches by normal means will be performed during the refueling outages.

3.2 BASES (cont'd)

crease to the Safety Limit. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only three percent of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the Safety Limit.

The RBM rod block function provides local protection of the core: i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

The refueling interlocks also operate one logic channel, and are required for safety only when the Mode Switch is in the Refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in

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TABLE 3.2-3

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
2	APRM Upscale (Flow Biased)	$S \leq (0.66W+42\%) \times \left[\frac{FRP}{MFLPD}\right]$	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	$S \leq$ 0.66W+39%(8)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	$\geq$2% of full scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$86.4% of full scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Startup Position	(3)	4 Inst. Channels	(1)
2 (4) (5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)

NOTES FOR TABLE 3.2-3

1. For the Startup and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in run mode, and

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3 (Cont'd)

the APRM and RBM rod blocks need not be operable in startup mode. From and after the time it is found that the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. From and after the time it is found that the first column cannot be met for both trip systems, the systems shall be tripped.

2. IRM downscale is bypassed when it is on its lowest range.

3. This function is bypassed when the count rate is $\geq$ 100 cps.

4. One of the four SRM inputs may be bypassed.

5. This SRM Function is bypassed when the IRM range switches are on range 8 or above.

6. The trip is bypassed when the reactor power is $\leq$ 30%.

7. This function is bypassed when the Mode Switch is placed in Run.

8. S = Rod Block Monitor Setting in percent of initial

   W = Loop recirculation flow in percent of rated (rated loop recirculation flow is 34.2×10^6 lb/hr).

3.3 (Cont'd)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

   a. Both RBM channels shall be operable, or

   b. Control rod withdrawal shall be blocked, or

   c. The operating power level shall be limited so that MCPR will remain above the Safety Limit assuming a single error that results in complete withdrawal of any single operable control rod.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

3.3 and 4.3 BASES (cont'd)

rods have been withdrawn (e.g. groups A_{12} and A_{34}), it is demonstrated that the Group Notch made for the control drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 20% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

4. The Source Range Monitor (SRM) System performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per sec assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transient cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e. MCPR limits as shown in specification 3.1.B). During use of such patterns, it is judged that testing of the RBM System prior to withdrawal of such rods to assure its operability will assure that improper withdraw does not occur. It is the responsibility of the Reactor Analyst to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the Plant Superintendent.

C. Scram Insertion Times

The Control Rod System is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. The limiting power transient is that

3.3 and 4.3 BASES (cont'd)

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-24129-1 Figures 1 and 2) with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than the Safety Limit.

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

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.5.1 through 3.5.6. If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI, RCIC, or Core Spray System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI, RCIC, and Core Spray shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested every month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

3.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \left(1 - \left[(\Delta P/P)_{max} \; (L/LT)\right]\right)$$

$LHGR_d$ = Design LHGR = G KW/ft.

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = N

LT = Total core length = 12 feet

L = Axial position above bottom of core

G = 18.5 KW/ft for 7x7 fuel bundles
= 13.4 KW/ft for 8x8 and 8x8R fuel bundles

N = 0.026 for 7x7 fuel bundles
= 0.022 for 8x8 and 8x8R fuel bundles

If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the LHGR is returned to within the prescribed limits.

4.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

3.5 BASES (cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, RCIC, and HPCI are not filled, a water hammer can develop in this piping when the pump(s) are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is required to be operable. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for technical specification purposes. However, if a water hammer were to occur, the system would still perform its design function.

H. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50 Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm 20^{\circ}F$ relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.5.1 through 3.5-6.

I. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq 25\%$ power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the ratio of local LHGR to average LHGR would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

Figure 3.5-5

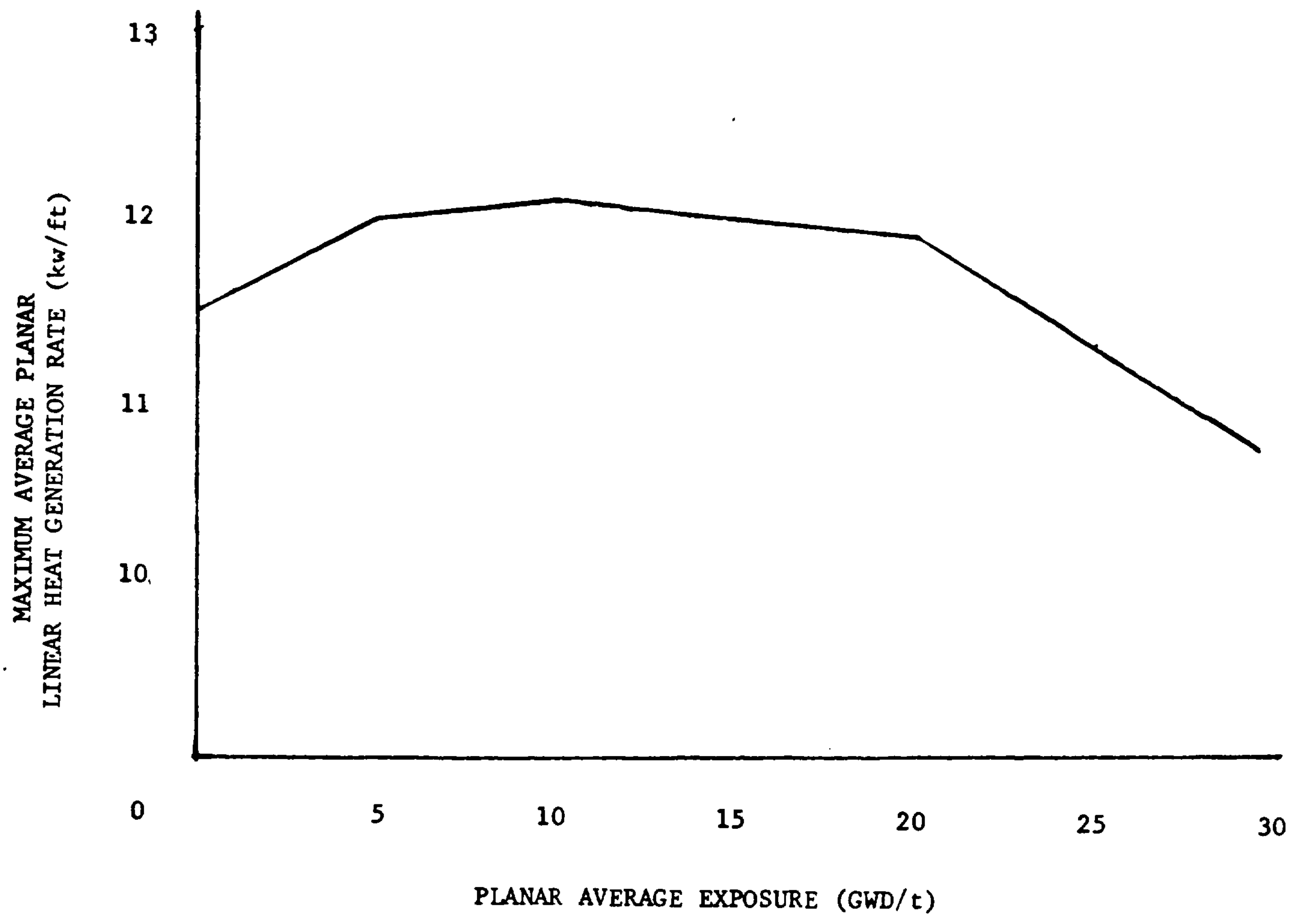

FIGURE 3.5-5 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

RELOAD 8DRB265L

REFERENCE
NEDO-24129
SECTION 14

FULL CORE DRILLED

Figure 3.5-6

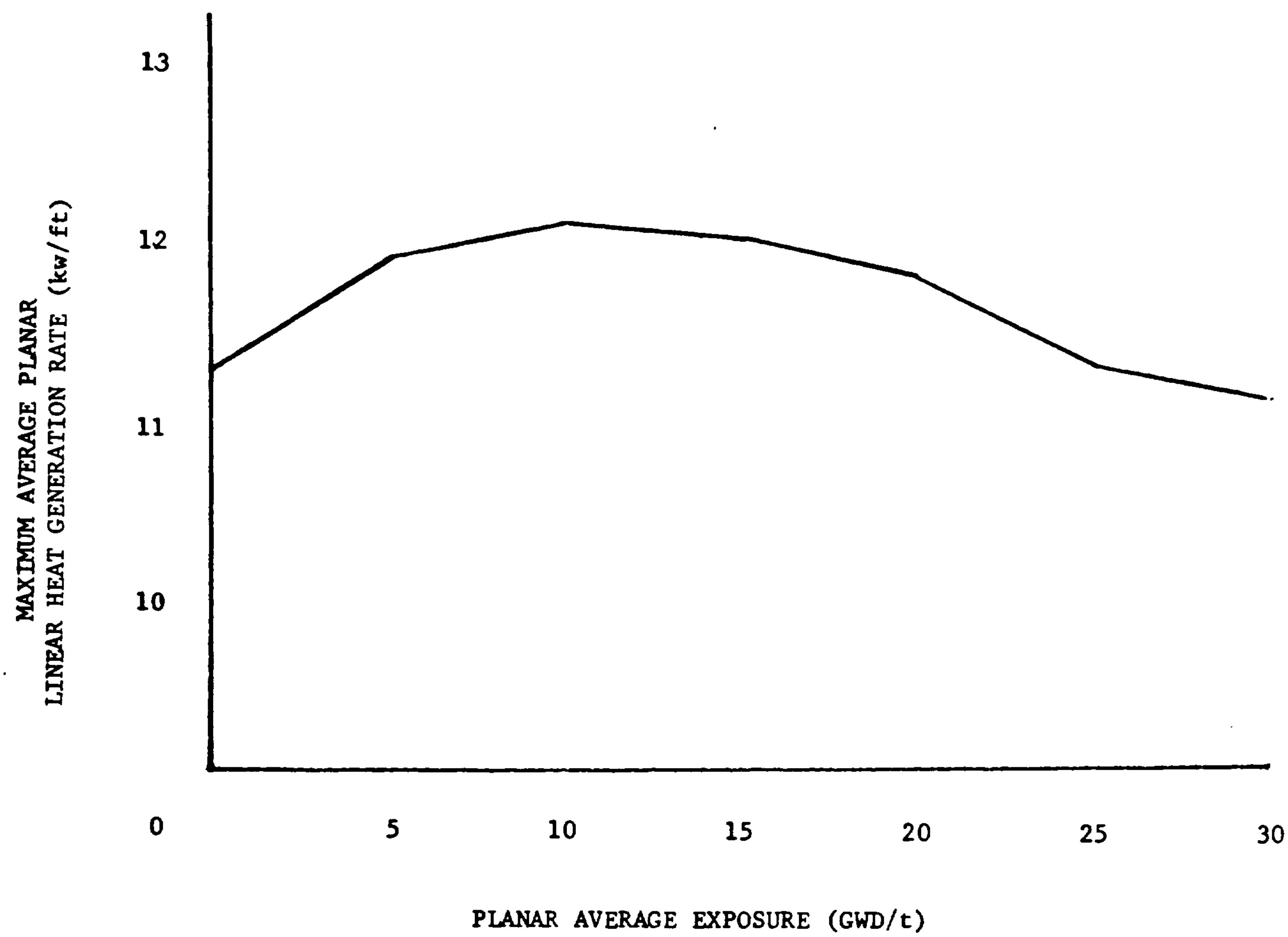

FIGURE 3.5-6 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

RELOAD 8DRB283

REFERENCE
NEDO-24129
SECTION 14

FULL CORE DRILLED

2. a. From and after the date that the safety valve function of one safety/relief valve is made or found to be inoperable, continued operation is permissible only during the succeeding 30 days unless such valve is sooner made operable.

   b. From and after the time that the safety valve function on two safety/relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 7 days unless such valves are sooner made operable.

3. If Specification 3.6.B.1 and 3.6.B.2 are not met, the reactor shall be placed in a cold condition within 24 hr.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Item B.2 above, provided that reactor coolant temperature is $\leq 212^{o}F$ and the reactor vessel is vented or the reactor vessel head is removed.

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the safety/relief valve bellows shall be continuously monitored.

   a. The bellows monitoring pressure switches shall be removed and bench checked once/operating cycle. Modified safety/relief valves with two-stage assemblies do not have a bellows arrangement and are, therefore, not subject to this requirement.

4. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

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.

The modified version of the safety/relief valves function with a direct-acting pilot arrangement with no integral bellows.

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. For those modified safety/relief valves with the direct-acting pilot arrangement, bellows failures and bellows related calibrations do not apply.

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

5.0 DESIGN FEATURES

5.1 SITE

A. The James A. FitzPatrick Nuclear Power Plant is located on the PASNY portion of the Nine Mile Point site, approximately 3,000 ft. east of the Nine Mile Point Nuclear Station. The NMP-JAF site is on Lake Ontario in Oswego County, New York, approximately 7 miles northeast of Oswego. The plant is located at coordinates north 4,819,545.012 m, east 386,968.945 m, on the Universal Transverse Mercator System.

B. The nearest point on the property line from the reactor building and any points of potential gaseous effluents, with the exception of the lake shoreline, is located at the northeast corner of the property. This distance is approximately 3,200 ft. and is the radius of the exclusion area as defined in 10 CFR 100.3.

5.2 REACTOR

A. The reactor core consists of not more than 560 fuel assemblies. For the current cycle three fuel types are present in the core: 7 x 7, 8 x 8, and 8 x 8R. These fuel types are described in Section 3.2 of the FSAR, NEDO-20360, and NEDO-24011, respectively. The 7 x 7 fuel has 49 fuel rods, the 8 x 8 fuel has 63 fuel rods and 1 water rod, and the 8 x 8R fuel has 62 fuel rods and 2 water rods.

B. The reactor core contains 137 cruciform-shaped control rods as described in Section 3.4 of the FSAR.

5.3 REACTOR PRESSURE VESSEL

The reactor pressure vessel is as described in Tables 4.2-1 and 4.2-2 of the FSAR, The applicable design codes are described in Section 4.2 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters and characteristics for the primary containment are given in Table 5.2-1 of the FSAR.

B. The secondary containment is as described in Section 5.3 and the applicable codes are as described in Section 12.4 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations are designed in accordance with standards set forth in Section 5.2 of the FSAR.

5.5 FUEL STORAGE

A. The new fuel storage facility is designed so that the Keff dry is <0.90 and flooded is <0.95 described in Section 9.2 of the FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 43 TO FACILITY 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 August 18, 1978[1] and supplemented by letters dated October 13, 1978,[2] November 16, 1978[3] and November 17, 1978,[4] the Power Authority of the State of New York (PASNY), the licensee, requested amendment to the Technical Specifications appended to Operating License DPR-59 for James A. FitzPatrick Nuclear Power Plant (JAFNPP). The proposed changes relate to the refueling of JAFNPP, for Cycle 3 operation. It involves: (1) the replacement of 136 exposed 7x7 fuel assemblies with a like number of fresh, two water rod, retrofit 8x8 fuel assemblies (8x8R) designed and fabricated by the General Electric Company (GE); (2) the raising of setpoints and regrouping of reactor coolant system safety/relief valves (SRV) for Mark I Containment Short Term Program; and (3) modifications to the APRM rod block and trip setpoint formulation and system. In support of this reload application, the licensee has submitted a supplemental reload licensing document[5] prepared by GE, and proposed Technical Specification changes.[1-4]

This reload is the first in which JAFNPP has incorporated the 8x8R fuel design. The description of the nuclear and mechanical design of the Reload 2 8x8R fuel and the exposed fuel designs of the initial core and Reload 1 is contained in GE's generic licensing topical report for BWR reloads.[6] Reference 6 also contains a complete set of references to GE's topical reports which describe GE's BWR reload analysis methods for the nuclear, mechanical, thermal-hydraulic, transient and accident calculations, and information on the applicability of these methods to cores with a mixture of different fuel designs. Portions of the plant-specific data, such as operating conditions and design parameters which are used in transient and accident calculations, have also been included in the topical report.

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Our safety evaluation[7] of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, as applied to mixed cores containing 7x7, 8x8, and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was expressed in the staff's evaluation[8] of Reference 9.

As part of our evaluation[7] of Reference 6, we found the cycle-independent input data for the reload transient and accident analyses to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 5, which follows the format and content of Appendix A of Reference 6.

As a result of the staff's generic evaluation of a substantial number of safety considerations on the use of 8x8R fuel in mixed core loadings with 8x8 and 7x7 fuel,[7] only a limited number of additional review items are included in this evaluation. These include the plant and cycle-specific input data and results and the LOCA-ECCS analysis results for the reload fuel design.

In letters dated June 7, 1978, July 31, 1978, August 18, 1978, August 25, 1978, September 28, 1978, and November 14, 1978, the licensee responded to a staff request for an interim assessment of the potential for and consequences of multiple-consecutive safety-relief valve (SRV) actuations following a reactor isolation transient, which was transmitted in a letter dated March 20, 1978. The licensee's assessment indicated that some form of corrective action would be necessary to satisfy the acceptance criteria specified by the staff. The licensee subsequently proposed to stagger the setpoints for the SRVs to limit the number of valves which could experience consecutive actuation following an isolation transient, as discussed in Section 4.0, herein. The reactor performance characteristics of this change are discussed in Sections 2.3.1 and 2.4 of this evaluation.

2.0 Evaluation

2.1 Nuclear Characteristics

For the upcoming cycle, 136 fresh 8x8R fuel bundles, will be loaded into the core (100 8DRB283 and 36 8DRB265L), replacing a like number of exposed 7x7 assemblies. The remainder of the 560 fuel assembly core will consist of the irradiated 7x7 and 8x8 fuel assemblies exposed during the first two fuel cycles. The reference core loading for Cycle 3 will result in eighth core symmetry, which is consistent with previous cycles.

The information provided in Section 6 of Reference 5 indicates that the fuel temperature and void dependent behavior of the reconstituted core is not significantly different from previous cycles. Additionally, scram effectiveness, as shown in Figures 2 and 3 of Reference 5, is also similar to earlier cycles. The 1.2%Δk/k calculated shutdown margin for the reconstituted core meets the requirement that the core be subcritical by at least 0.38%Δk/k in the most reactive operating state with the single most reactive control rod fully withdrawn and all other rods fully inserted. Finally, Reference 5 indicates that a boron concentration of 600 ppm in the moderator will provide a shutdown margin of at least 3.0%Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System.

2.2 Thermal-Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 7, for BWR cores which reload with GE's retrofit 8x8R fuel, the allowable minimum critical power ratio (MCPR), from either core-wide or localized abnormal operational transients, is equal to 1.07. With this MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee for Cycle 3 represents a .01 increase from the 1.06 SLMCPR applicable during Cycle 2. The basis for the revised safety limit is addressed in Reference 6, while our generic approval of the new limit is given in Reference 7.

2.2.2 Operating Limit MCPR

Various transient events will reduce the MCPR from its normal operating value. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed by the licensee to determine which event results in the largest reduction in the minimum critical power ratio. Each of the events has been analyzed for each of the several fuel types (i.e., 7x7, 8x8, 8x8R) and at several exposure intervals through the full range of exposure for the cycle.

The methods used for these calculations, including cycle-independent initial conditions and transient input parameters are described in Reference 6. Our acceptance of the values used and related transient analysis methods appear in Reference 7. Supplementary cycle-dependent initial conditions and transient input parameters used in the analysis appear in the table in Section 6 and 7 of Reference 5. Our evaluation of the methods used to develop these supplementary transient input values have already been addressed and appear in Reference 7. The overall transient methodology, including cycle-independent transient analysis inputs, provides an adequately conservative basis for the determination of transient MCPRs. The transient events analyzed were load rejection without bypass, turbine trip without bypass, feedwater controller failure, loss of feedwater heating, and control rod withdrawal error.

Based on our review, the limiting abnormal operational transients and associated MCPRs are as shown in Section 11 of Reference 5.

Thus, when the reactor is operated in accordance with the proposed operating limit MCPRs, the 1.07 SLMCPR will not be violated in the event of the most severe abnormal operational transient. This is acceptable to the staff.

2.2.3 Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawal error event was also analyzed by the licensee using methods acceptable to the staff to determine the maximum linear heat generation rates (LHGR). The results show that the fuel type and exposure dependent safety limit LHGRs, given in Table 2-3 of Reference 6 will not be violated should this event occur.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License, to implement the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading..."the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions.

The licensee has reevaluated the adequacy of ECCS performance in connection with the new reload fuel design, using methods previously approved by the staff. The results of these plant-specific analyses are given in Reference 5.

The licensee has also presented the results of a small break LOCA analysis with the revised SRV setpoints[13] per Section 4.0 herein. These results indicate no significant change in peak cladding temperature from the previously reported value of 1285°F. The SRV setpoint change does not significantly affect the remaining ECCS performance analyses.

We have reviewed the information submitted by the licensee and conclude that all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 will be met when the reactor is operated in accordance with the MAPLHGR versus Average Planar Exposure values given in Section 14 of Reference 5.

2.3.2 Control Rod Drop Accident

For the worst case control rod drop accident (CRDA) during hot or cold startup conditions, the key plant-specific nuclear characteristics are within the bounds of those used in the bounding CRDA analysis given in Reference 6 except for cold startup reactivity shape. The licensee has stated that this variation from the bounding point (early in the rod drop) does not affect the outcome of the CRDA. We have reviewed this and agree with the licensee. Since the bounding analysis showed that the peak fuel enthalpy does not exceed the 280 cal/gm fuel enthalpy design limit, the peak fuel enthalpy associated with a CRDA from hot or cold startup condition will also be within the 280 cal/gm design limit.

2.3.3 Fuel Loading Error (FLE)

Guidance for the evaluation of the FLE is given in the Standard Review Plan (SRP) Section 15.4.7. This section requires that either the FLE must be detectable by available nuclear instrumentation and hence remediable prior to fuel failure or the consequences of the most severe FLE must be shown to remain a small fraction of 10 CFR 100 guidelines. In a BWR, the former of these criteria cannot be satisfied because the current instrumentation does not cover all fuel locations. In consideration of the latter criterion, we currently find it sufficient if the worst case FLE does not violate the safety limit MCPR which precludes significant fuel damage and thereby meets the small fraction of 10 CFR 100 criteria. An analysis of the most severe mislocated and misoriented fuel loading errors with GE's methodology,[10,11] which has been found acceptable as modified by our evaluation,[12] shows that this event will not cause a violation of the safety limit MCPR. On this bases we find the FLE to meet the criteria of the SRP and, therefore, to be acceptable.

2.4 Overpressure Analysis

The licensee has reanalyzed the limiting pressurization transient to demonstrate that the ASME Boiler and Pressure Vessel Code requirements are met. The methods used for this analysis, when modified to account for one failed safety valve, have been previously approved by the staff. The acceptance criteria for this event is that the calculated peak transient pressure not exceed 110% of design pressure, i.e., 1375 psig.

Reactor coolant system safety/relief valve (SRV) setpoints have been raised and regrouped to avoid spurious opening, per Section 4.0, herein. The safety analysis presented in Reference 13 uses three new valve setpoint groups: 2 valves at 1090 psig, 2 valves at 1105 psig and 7 valves at 1140 psig, as presented on page 27 of the proposed specifications. Allowable setpoint error remains at ±1%.

The overpressure protection analysis presented in Reference 13 indicates that in the case of the most severe isolation event (closure of all main steamline isolation valves with failure of the direct scram on position and reliance instead on the indirect scram on high flux, evaluated at full power end of Cycle 3 conditions), peak pressure rise at the bottom of the vessel reaches 1264 psig. This results in a 111 psi margin below the vessel ASME code limit of 1375 psig. This analysis shows that the peak pressure at the bottom of the reactor vessel is less than the 110% criteria for worst case end-of-cycle conditions, even when the effects of one failed safety valve are considered.

2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed with the methods described in Reference 6. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state (corresponding to the intersection of the natural circulation curve and 105% rod line on the power-flow map) are below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. Although a final test report has not as yet been received by the staff for review, it is expected that the test results will aid considerably in resolving the staff concerns.

For the previous operating cycle, the staff, as an interim measure, added a requirement to the Technical Specifications which restricted planned operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins for the current cycle so that the decay ratio is <1.0 in all operating modes. On the basis of the foregoing, the staff considers the plant thermal-hydraulic stability characteristics to be acceptable.

3.0 Physics Startup Testing

The licensee will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during the cycle. The tests will check that the core is loaded as intended, that the incore monitoring system is functioning as expected, and that the process computer has been reprogrammed to properly reflect changes associated with the reload.

The licensee has agreed to provide a written report of the startup tests within 45 days. This test program is acceptable.

4.0 Multiple-Consecutive Safety/Relief Valve Actuations

Following a reactor isolation transient, multiple-consecutive SRV actuations could occur which would result in increased loadings on the suppression chamber and its support structures. In a letter dated March 20, 1978, the staff requested that the licensee perform an interim assessment of the containment response to a multiple-consecutive SRV actuation to justify deferment of this issue until it is ultimately resolved as part of the Mark I Containment Long Term Program. In that letter, the assumptions and acceptance criteria for this assessment were set forth, based on data from Monticello in-plant SRV tests.

The licensee's assessment indicated that some form of corrective action would be necessary to satisfy the acceptance criteria. The licensee subsequently proposed to stagger the SRV setpoints to limit the number of values which could experience multiple-consecutive actuations following an isolation transient. We have reviewed the analyses presented by the licensee and determined that the assessment has been performed in accordance with the staff's requirements. We conclude that the SRV setpoints proposed by the licensee will assure that the analysis of the containment structure for the effects of multiple-consecutive relief valve actuations satisfies the structural acceptance criteria set forth in the Mark I Short Term Program. Therefore, we conclude that this issue can be deferred for the FitzPatrick plant until its ultimate resolution in the Mark I Containment Long Term Program.

5.0 Modifications to APRM Rod Block and Trip Setpoint Formulation and System

5.1 Modifications to the APRM Flow-Biased Flux Scram and Rod Block Setpoints

The equations given in the current Technical Specifications for the APRM flux scram setpoint and the APRM rod block setpoint have been changed. The proposed changes replace the trip reduction factor and criterion with a new reduction factor and a new criterion which are defined by quantities which are directly available from the process computer. The present specification requires that the slope and intercept of the flow biased scram and rod block lines be reduced by the factor PF/MTPF (PF is the design total peaking factor, MTPF is the maximum total peaking factor) whenever the maximum total peaking factor is greater than the design total peaking factor. The proposed specifications require that the slope and intercept of the flow biased scram and rod block lines be reduced by the factor FRP/MFLPD whenever the maximum fraction of limiting power density is greater than the fraction of rated power. In the above, FRP is the fraction of rated power and MFLPD is the maximum fraction limiting power density. The limiting power densities are 13.4 KW/ft for 8x8 and 8x8R bundles, and 18.5 KW/ft for 7x7 bundles. This is only a change in the formulation of these setpoints and algebraically produces the same setpoint. This formulation is currently used in the Browns Ferry Technical Specifications. The change is desired to make the administrative control of this setpoint easier. On these bases, we find the change acceptable.

5.2 Modifications to the RPS for Thermal Power Monitor Installation

New APRM scram trip logic will be installed during the refueling outage. The new logic will reduce the number of spurious high flux scrams. Such scrams are the result of momentary neutron flux spikes caused by small changes in recirculation system flow and small pressure disturbances during turbine stop valve and control valve testing and are not desirable in that they impose an unnecessary transient on the reactor core which may affect fuel performance.

The existing flow referenced scram utilizes APRM neutron flux measurements to estimate the peak heat flux level in the core. This is satisfactory for steady-state operation, but over-predicts the fuel heat flux level during power increase events. During such events, the neutron flux leads the heat flux because of the fuel time constant.

Therefore, neutron flux trip levels are reached before the reactor heat flux has actually increased to the scram level. While this anticipatory response in the APRM scram is desirable to protect the core during abnormal operational transients or accidents, it may result in spurious scrams for momentary neutron flux spikes.

Many of these spurious scrams will be avoided by the installation of the Thermal Power Simulator and an APRM Simulated Thermal Power Trip Unit (hereafter called the Thermal Power Monitor). The unit provides a signal which is representative of the heat flux during a transient. Utilizing the APRM neutron flux signal, an output signal can be obtained which closely approximates the heat flux during a transient or steady state condition. This is accomplished by a filtering network with a time constant which is representative of the fuel dynamics.

At present, Brunswick Units 1 and 2 are the only domestic BWR plants which are operating with the new APRM scram trip logic. This logic was an integral part of the APRM scram trip system when these plants were initially licensed (see Section 7.5.5, Average Power Range Monitor Subsystem, in the Brunswick Units 1 and 2 FSAR).

Field experience from these plants has shown that spurious scrams from recirculation system excursions have been reduced by 50 to 75% due to this modification. A similar reduction on spurious scrams is expected when the new APRM scram trip logic is installed in the FitzPatrick plant.

Analyses for Fitzpatrick Cycle 3 have demonstrated that with only the 120% trip setting, none of the abnormal operational transients analyzed violates the fuel cladding integrity safety limit. Therefore, the use of the flow referenced trip setpoint, with the fixed setpoint as backup, provides adequate thermal margins for fuel cladding integrity.

On these bases we find the proposed modification acceptable.

6.0 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 Section 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.

7.0 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: November 22, 1978

References

1. PASNY letter (Early) to USNRC (Ippolito) "James A. FitzPatrick Nuclear Power Plant Reload 2 Licensing Submittal for Cycle 3 Operation, Docket No. 50-333" dated August 18, 1978.

2. PASNY letter (Early) to USNRC (Ippolito) "James A. FitzPatrick Nuclear Power Plant Reload 2 Licensing Supplement for Cycle 3 Operation, Docket No. 50-333" dated October 13, 1978.

3. PASNY LETTER (Early) to USNRC (Ippolito) "James A. FitzPatrick Nuclear Power Plant Rotated Bundle Loading Error Event Analysis, Docket No. 50-333" dated November 16, 1978.

4. PASNY letter (Early) to USNRC (Ippolito) dated November 17, 1978.

5. "Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant for Reload 2," NEDO-2429, June 1978.

6. "Generic Reload Fuel Application," General Electric Report, NEDE-24011-P-3, dated March 1978.

7. USNRC letter (Eisenhut) to General Electric (Gridley) dated May 12, 1978, transmitting "Safety Evaluation for the General Electric Topical Report, 'Generic Reload Fuel Application,' (NEDE-24011-P)."

8. "Status Report on the Licensing Topical Report, General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by the Division of Technical Review, Office of Nuclear Reactor Regulation, USNRC, April 1975.

9. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1, Supplement 4, April 1, 1976.

10. GE letter (Engle) to NRC (Eisenhut), "Fuel Assembly Loading Error" dated June 1, 1977.

11. GE letter (Engle) to NRC (Eisenhut) dated November 30, 1977.

12. NRC letter (Eisenhut) to GE (Engle) dated May 8, 1978.

13. Zull, L. M., "Raised Safety/Relief Valve Setpoint Reanalysis for the James A. FitzPatrick Nuclear Power Plant for Reload No. 2," NEDO-24129-1, Supplement 1, September 1978.

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. 43 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 by: (1) revision of the specification as a result of Safety Relief Valve regrouping and setpoint changes; (2) revision of the specification to reflect reactor refueling using the General Electric 8x8R fuel; and (3) revisions to reflect miscellaneous minor changes to correct editorial errors in the current specifications.

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 Section 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 August 18, 1978, (2) Amendment No. 43 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 22nd day of November 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors