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September 16, 1977

Docket No. 50-333

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant (the facility). The amendment consists of changes to the Technical Specifications in response to your application for amendment submitted by letter dated May 16, 1977, as supplemented, and in partial response to your application for amendment submitted by letter dated July 25, 1977, as supplemented.

The amendment authorizes operation of the facility with: (1) 8X8 reload fuel bundles with 100 mil channels, (2) holes drilled in the lower tie plate of all reload fuel bundles and all first cycle fuel remaining in the core after refueling, (3) independent power supplies for the Low Pressure Coolant Injection System Motor Operated Valves, (4) the valve of the control rod drive hydraulic return line placed in the closed position and (5) limiting Maximum Average Planar Linear Heat Generation Rates as determined by a reevaluation of the Emergency Core Cooling System (ECCS) cooling performance. Effective upon issuance of this amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-59, is terminated.

Cont. 1

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Changes and additions to the Technical Specifications proposed in your applications dated May 16, 1977 and July 25, 1977, were necessary. These changes have been discussed with and agreed to by your staff.

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

Sincerely,

Original Signed by

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

Enclosures:
1. Amendment No. 30
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

C-PSB-OT:DOR WButler 9/14/77 | C-RSB-OT:DOR RBaer 9/14/77 | C-EEB-OT:DOR BGrimes 9/14/77

see comments which

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| DATE | 9/13/77 | 9/14/77 | 9/16/77 | 9/16/77 | 9/16/77 | |

Power Authority of the State
of New York

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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. 30
License No. DPR-59

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

   A. The applications for amendment by the Power Authority of the State of New York (the licensee) sworn to, as supplemented, comply 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 applications, 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. 30 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 16, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 30

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. No changes were made on overleaf pages which are identified below by an asterisk:

| Remove Pages | Insert Pages |
|---|---|
| 6 & 7 | 6 & 7 |
| 9 & 10 | 9 & 10 |
| 12 & 13 | 12 & 13 |
| 15 - 19 | 15 - 19 |
| 29 - 31 | 29 - 31 |
| 35 | 35 |
| 41 | 41 |
| 43 | 43 |
| 58 | 58 |
| 72 | 72 |
| 74 | 74 |
| 93a - 96 | 93a - 96 |
| 100 - 103 | 100 - 103 |
| 107* & 108 | 107* & 108 |
| 123 & 124 | 123 - 124a |
| 130 | 130 |
| 134 & 135 | 134 - 135b |
| 145 | 145 |
| 215 | 215 |
| - | 222a - 222c |
| 223* & 224 | 223* & 224 |
| 225* & 226 | 225* & 226 |
| 245 & 246* | 245 & 246* |

1.0 (cont'd)

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. Total Peaking Factor - The total peaking factor shall be the ratio of local LHGR divided by the average LHGR for any specific location on a fuel rod.

3. 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.

## 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 1.06 shall constitute violation of the fuel cladding integrity 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 $\leq$ 120/125 of full scale.

1.1 (Cont'd)

1.1.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)

2.1.A.1.c. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM 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 $34.2 \times 10^6$ lb/hr)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design values, the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)\ \frac{P.F.}{MTPF}$$

where:

P.F. = Design value of total peaking factor
= 2.60 (7x7)
= 2.42 (8x8)

MTPF = The value of the existing maximum total peaking factor

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

1.1 (Cont'd)

2.1 (Cont'd)

2.1.A.1.d. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

$$S_{RB} \leq 0.66\ W + 42\%$$

where:

$S_{RB}$=Rod block setting in percent of rated thermal power (2436 MWt)

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

In the event of operation with a maximum total peaking factor (MTPF) greater than the design values, the setting shall be modified as follows:

$$S_{RB} \leq (0.66\ W + 42\%)\ \frac{P.F.}{MTPF}$$

where:

P.F. = Design value of total peaking factor
= 2.60 (7x7)
= 2.42 (8x8)

MTFP = The value of the existing maximum total peaking factor

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 1.06. MCPR > 1.06 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 1.06) 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 1.06 is derived from a detailed statistical analysis considering all of the

1.1 BASES (Cont'd.)

*uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference I. 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 condition of MCPR = 1.06 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$^{o}$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 (MCPR = 1.06) operation is constrained to a maximum LHGR = 18.5 kw/ft for 7 x 7 fuel and 13.4 kw/ft for 8 x 8 fuel. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 2.60 for 7 x 7 fuel and 2.42 for 8 x 8 fuel. For the case of the MTPF exceeding the above operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram and rod block settings as required by Specifications 2.1.A.1.c and 2.1.A.1.d.*

B. *Core Thermal Power Limit (Reactor Pressure < 785 psig)*

*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 28x10$^{3}$ 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 28x10$^{3}$ 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

## 2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Fitzpatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2535 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 is the licensed maximum power level of Fitzpatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity, coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. Active coolant flow is equal to 88% of total core flow. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion.

The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The MCPR operating limits of specifications 3.1.B are conservatively assumed to exist prior to initiation of the transients.

*This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.*

*Steady-state operation without forced recirculation will not be permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.*

*In summary:*

*i. The abnormal operational transients were analyzed to a power level of 2535 MWt.*

*ii. The licensed maximum power level is 2436 MWt.*

*iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.*

*iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.*

*A. Trip Settings*

*The bases for individual trip settings are discussed in the following paragraphs.*

*1. Neutron Flux Trip Settings*

*a. IRM Flux Scram Trip Setting*

*The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.*

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 1.06. 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 Start & 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 average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2436 MWt). Because fission chambers provide the basic input signals, the APRM

Amendment No. ~~14~~, ~~21~~, 30

2.1 BASES (Cont'd.)

system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin. An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.c, when the maximum total peaking factor is greater than the design values.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.06 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 1.06. 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 relationship; 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

2.1 BASES (Cont'd)

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 total peaking factor exceeds the design value shown in surveillance requirement 2.1.A.1.c thus preserving the APRM rod block safety margin.

2. Reactor Water Low Level Scram Trip Setting (LLI)

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 $\leq$ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 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 by passed 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

1.2 and 2.2 BASES

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

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

The analysis in NEDO-21619 Section 6.3.4 shows that the main steam isolation valve transient, when direct scram is ignored, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is at least 105 psig above the peak pressure produced by the event above. Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected (1,375 psig) overpressure transients. Figure 6-11 of NEDO-21619 presents the curve produced by this analyses. Reactor pressure is continuously indicated in the control room during operation.

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

## 3.1 LIMITING CONDITIONS FOR OPERATION

## 3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the Reactor Protection System.

Specification:

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

B. Minimum Critical Power Ratio (MCPR)

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

| FUEL TYPE | MCPR OPERATING LIMIT FOR INCREMENTAL CYCLE 2 CORE AVERAGE EXPOSURE | |
|---|---|---|
| | BOC 2 to 2000 MWd/t before EOC 2 | 2000 MWd/t before EOC 2 to EOC 2 |
| 7 x 7 | 1.22 | 1.28 |
| 8 x 8 | 1.20 | 1.36 |

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

## 4.1 SURVEILLANCE REQUIREMENTS

## 4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

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

3.1 (cont'd)

reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is 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 $\geq$ 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.

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

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

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis for cycle 2 (NEDO 21619) 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.

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 High Flux | $S_N \leq (P.F/MTPF) x S_O$ (12) | | | X | 6 Instrument Channels | A or B |
| 2 | APRM Inoperative | (10) | X | X | X | 6 Instrument Channels | A or B |
| 2 | APRM Downscale | ≥2.5 indicated on scale (9) | | | X | 6 Instrument Channels | A or B |
| 2 | APRM High Flux in Startup | ≤15% power | X | X | | 6 Instrument Channels | A |
| 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 |



Amendment No. ~~19~~, 90

TABLE 3.1-1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES FOR TABLE 3.1-1 (cont'd)

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

   A. Mode Switch in Shutdown

   B. Manual scram

   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 total peaking factor (MTPF) greater than the design value.

    where:

    P.F. = Design value of total peaking factor
    = 2.60 (7x7)
    = 2.42 (8x8)
    MTPF = The value of the existing maximum total peaking factor
    $S_o$ = 0.66 W + 54%
    W = Loop Recirculation flow in percent of rated (rated is $34.2 \times 10^6$ lb/hr)
    $S_n$ = Scram setting in percent of initial

crease to 1.06. 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 > 1.06.

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

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 ≤ (MTPF/PF)S | 6 Inst. Channels | (1) |
| 2 | APRM Upscale (Start-up Mode) | ≤ 12% | 6 Inst. Channels | (1) |
| 2 | APRM Downscale | ≥ 2.5 indicated on scale | 6 Inst. Channels | (1) |
| 1 (6) | Rod Block Monitor (Flow Biased) | S ≤ 0.66W+39% (8) | 2 Inst. Channels | (1) |
| 1 (6) | Rod Block Monitor Downscale | ≥ 2.5 indicated on scale | 2 Inst. Channels | (1) |
| 3 | IRM Downscale (2) | ≥ 2.5 indicated on scale | 8 Inst. Channels | (1) |
| 3 | IRM Detector not in Startup Position | (7) | 8 Inst. Channels | (1) |
| 3 | IRM Upscale | ≤ 108 indicated on scale | 8 Inst. Channels | (1) |
| 2 (4) | SRM Detector not in Startup Position | (3) | 4 Inst. Channels | (1) |
| 2 (4) (5) | SRM Upscale | ≤ $10^5$ 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 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 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.

TABLE 3.2-4

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

| Minimum No. of Operable Instrument Channels (1) | Trip Function | Trip Level Setting | Total Number of Instrument Channels Provided by Design for Both Channels | Action (2) |
|---|---|---|---|---|
| 1 | Refuel Area Exhaust Monitor | $\leq 2.7 \times 10^5$ cpm (5) | 2 Inst. Channels | A or B |
| 1 | Reactor Building Area Exhaust Monitors | $\leq 2.7 \times 10^5$ cpm (5) | 2 Inst. Channels | B |
| 1 | Off-gas Radiation Monitors | $\leq$1.0 ci/sec(3) | 2 Inst. Channels | C.1, C.2<br>C.3<br>C.3 |
| 1 | Turbine Bldg. Exhaust Monitors | $\leq 1.8 \times 10^5$ cpm (5) | 2 Inst. Channels | C |
| 1 | Radwaste Bldg. Exhaust Monitor | $\leq 6.7 \times 10^5$ cpm (5) | 2 Inst. Channels | C |
| 1 | Main Control Room Ventilation Monitor | $\leq 4 \times 10^3$ cpm (6) | 1 Inst. Channels | D |
| 2 | Mechanical Vacuum Pump Isolation | $\leq$3 times normal full power background | 4 Inst. Channels | E |
| 1 | Liquid Radwaste Discharge Monitor | (4) | 1 Inst. Channel | F |

NOTES FOR TABLE 3.2-4

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

2. Action

   A. Cease operation of the refueling equipment.
   B. Isolate secondary containment and start the Standby Gas Treatment System.
   C.1 If radiation level exceeds 1.0 ci/sec (30 min. decay level), the off gas isolation valve shall close within one minute.
   C.2 If radiation level exceeds 0.3 ci/sec (30 min. decay level), for a period greater than 15 consecutive minutes, reactor shutdown shall be initiated immediately and the reactor placed in a cold condition within 24 hours.
   C.3 Refer to Specification 2.3.B.4 of the Environmental Technical Specifications.
   D. Control Room isolation is manually initiated.
   E. Uses same sensors as Primary Containment Isolation on high main steam line radiation. Table 3.2-1.
   F. Refer to Environmental Technical Specification 2.3.A.3.

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

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

5. Conversion factor is $9.0 \times 10^7$ cpm = 1 μci/cc

6. Conversion factor is $8.15 \times 10^7$ cpm = 1 μci/cc

Amendment No. ~~27~~, 30

d. Control rod withdrawal sequence shall be established such that the drop of any in sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm.

e. If Specifications 3.3.B.3a through c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20 percent rated thermal power, it shall be brought to a shutdown condition immediately.

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 1.06. assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 (cont'd)

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 (cont'd)

6. During initial fuel loading or subsequent refuelings, the restraints imposed by Rod Sequence Control System groups $A_{12}$ and $A_{34}$, $B_{12}$ and $B_{34}$ may be bypassed to perform the required shutdown margin demonstration.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

| % Inserted From Fully Withdrawn | Avg. Scram Insertion Times (sec) |
|---|---|
| 5 | 0.375 |
| 20 | 0.90 |
| 50 | 2.0 |
| 90 | 3.50 |

4.3 (cont'd)

6. Prior to control rod withdrawal for startup or during refueling, verify the conformance to Specification 3.3.A.2.d before a rod may be bypassed in the Rod Sequence Control System.

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 950 psig (with saturation temperature). This testing shall be completed prior to exceeding 40% power. Below 20% power, only rods in those sequences ($A_{12}$ and $A_{34}$ or $B_{12}$ and $B_{34}$) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. During all scram time testing below 20% power the RWM shall be operable.

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

| % Inserted From Fully Withdrawn | Avg. Scram Inserton Times (sec) |
|---|---|
| 5 | 0.398 |
| 20 | 0.954 |
| 50 | 2.120 |
| 90 | 3.71 |

2. At 8-week intervals, 15 percent of the operable control rod drives shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the RSCS and the Rod Worth Minimizer (RWM).

2. The control rod housing support restricts the outward movement of a control rod to less than 3 in. in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the Primary Coolant System. The design basis is given in subsection 3.8.2 of the FSAR, and the safety evaluation is given in subsection 3.8.4. This support is not required if the Reactor Coolant System is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to pre-specified sequences. These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal/gm. An enthalpy of 280 cal/gm is well below the level at which rapid fuel dispersal could occur (i.e. 425 cal/gm/). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections III6.6, VIII7.4.5 and XIV6.2 of the FSAR and NEDO-10527 including supplements 1 and 2 to NEDO-10527.

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram drop limit. In this range, the RWM and RSCS constrain the control rod sequence and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviance from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences which limit the maximal reactivity worth of control rods , in the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the NPC can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural control to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.3.3.5 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of ± 2% of full power, the nominal instrument setting is 22% of rated power. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set manually at 30% of rated power to be consistent with the RSCS setting.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, $A_{12}$ and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the $A_{12}$ group is exclusive. By bypassing to full-out all $A_{12}$ rods, selecting $A_{34}$ and attempting to withdraw, by one notch, a rod or all rods in group B, the $A_{34}$ group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control 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 withdrawal 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 1.06 The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (NEDO- 21619 Figures 6-6.1 & 6-6.2) with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than 1.06

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

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

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

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

D. Reactivity Anomalies

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

C. Sodium Pentaborate Solution

The standby liquid control solution tank shall contain a boron bearing solution that satisfies the volume-concentration requirements of Fig. 3.4-1 at all times when the Standby Liquid Control System is required to be operable and the solution temperature including that in the pump suction piping shall not be less than the temperature presented in Fig. 3.4-2. Tank heaters shall be operable whenever the SLCS is required in order to maintain solution temperature in accordance with Fig. 3.4-2.

D. If specifications 3.4.A through C are not met, the reactor shall be in the cold condition within 24 hours.

C. Sodium Pentaborate Solution

The availability of the proper boron bearing solution shall be verified by performance of the following tests:

1. At least once per month –

   Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or boron is added or if the solution temperature drops below the limits specified by Fig. 3.4-2.

2. At least once per day –

   Solution volume and the solution temperature shall be checked.

3. At least once per operating cycle –

   The temperature and level elements shall be calibrated.

3.4 and 4.4 BASES

A. Normal Operation

The design objective of the Standby Liquid Control System is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Standby Liquid Control System is designed to inject a quantity of boron which produces a concentration of 600 ppm of boron in the reactor core in less than 125 min. Six hundred ppm boron concentration in the reactor core is required to bring the reactor from full power to a 3.0 percent $\Delta K$ subcritical condition considering the hot to cold reactivity swing, decay of xenon poisoning, and an additional margin (25 percent) for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 2,500 gal. of solution having a 17 percent sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (125 min) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required pumping rate of 39 gal per min, the maximum storage volume of the boron solution is established as 4,780 gal.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

The only practical time to test the Standby Liquid Control System is during a refueling outage and by initiation from local stations. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of more than once each refueling outage unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the charges are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the Standby Liquid Control System protect the system piping and positive displacement pumps, which are nominally designed for 1,500 psig,

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, 3.5.2 and 3.5.3. If at any time 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, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is 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 pressure switches which monitor the Core Spray and LPCI discharge lines to ensure that they are full shall be functionally tested every month and calibrated every three months.

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 ≥ 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:

$$LHGR_{max} \leq LHGR_d \quad (1 - \{\Delta P/P\}_{max} \quad (L/LT)\})$$

$LHGR_d$ = 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 fuel bundles

N = 0.026 for 7x7 fuel bundles
= .022 for 8x8 fuel bundles

If at any time 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 (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

**I. Linear Heat Generation Rate (LHGR)**

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

J. Thermal Hydraulic Stability

1. When the reactor mode switch is in STARTUP or RUN, the reactor shall not be operated in natural circulation mode.

2. With two recirculation loops out of service, action shall be initiated immediately to bring the reactor to hot shutdown within twelve hours. If forced recirculation is re-initiated within the first six hours of the 12 hours specified above, the conditions as described in 3.6.H.3 shall control.

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 the 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 ±20° 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, 3.5.2 and 3.5.3.

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 ≥ 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 MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

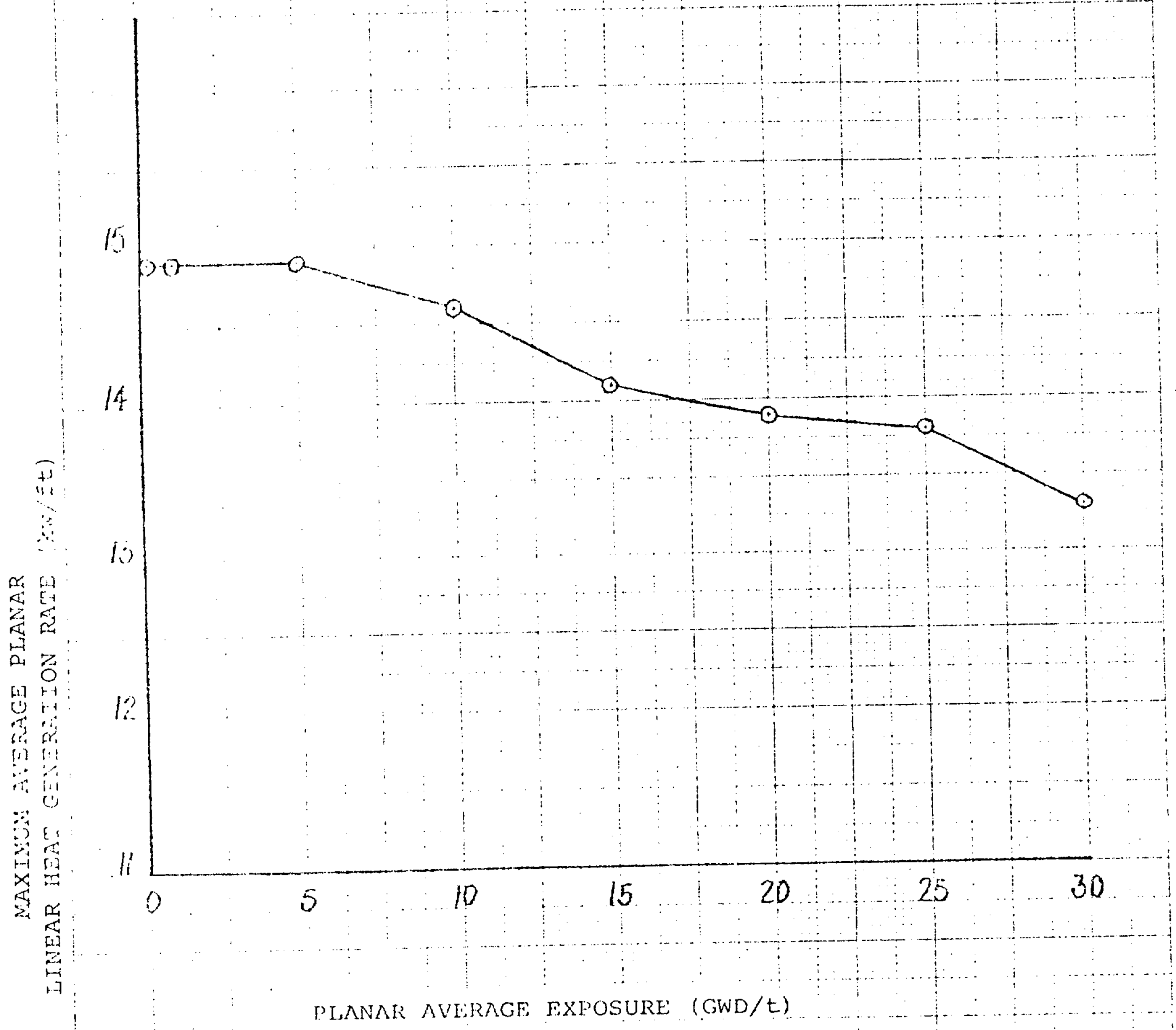


FIGURE 3.5-1 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

INITIAL CORE TYPE 2

FULL CORE DRILLED

REFERENCE
NEDO-21662-2
TABLE 5A

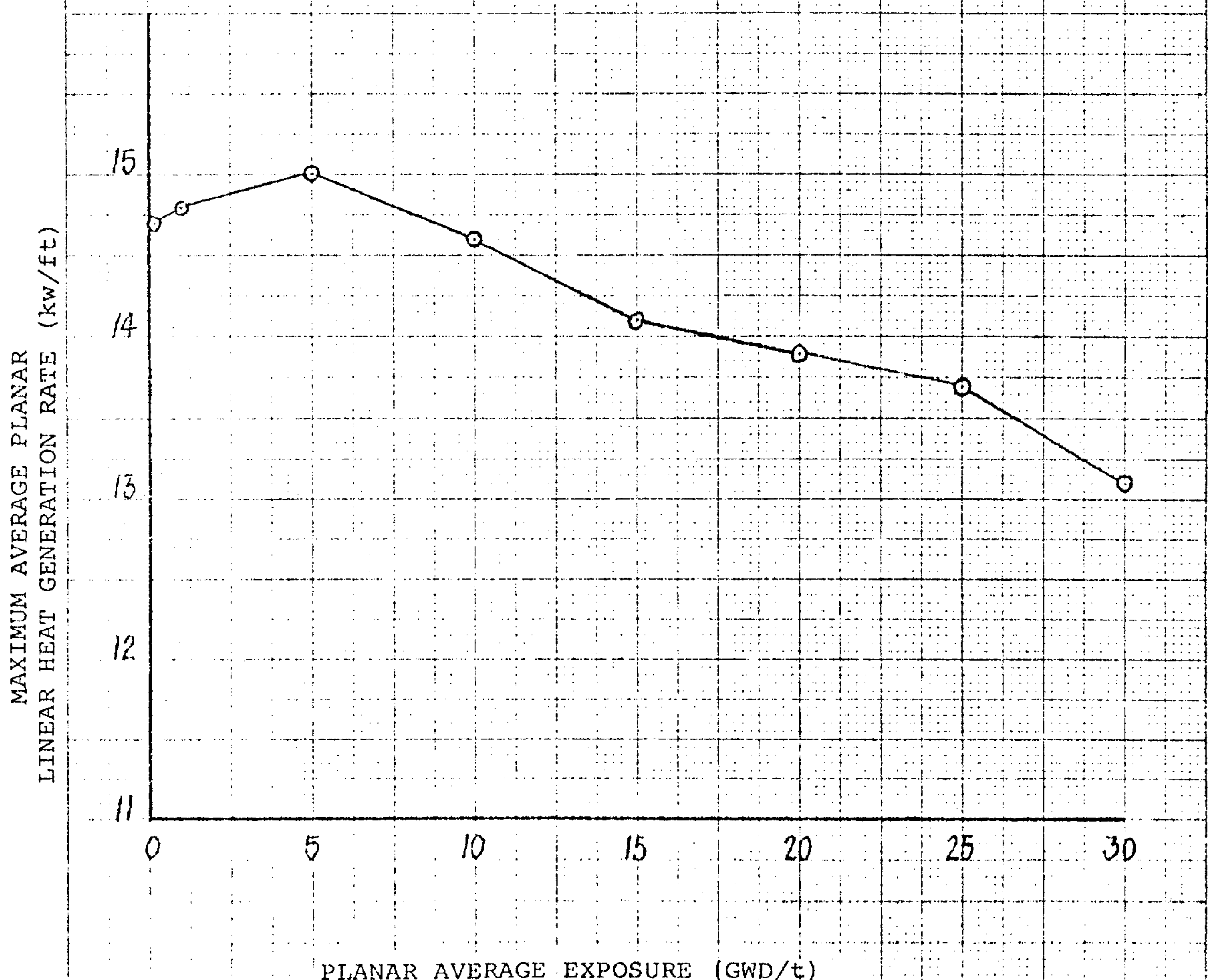


FIGURE 3.5-2 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

(MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

INITIAL CORE TYPE 3

FULL CORE DRILLED

REFERENCE
NEDO-21662-2
TABLE 5B

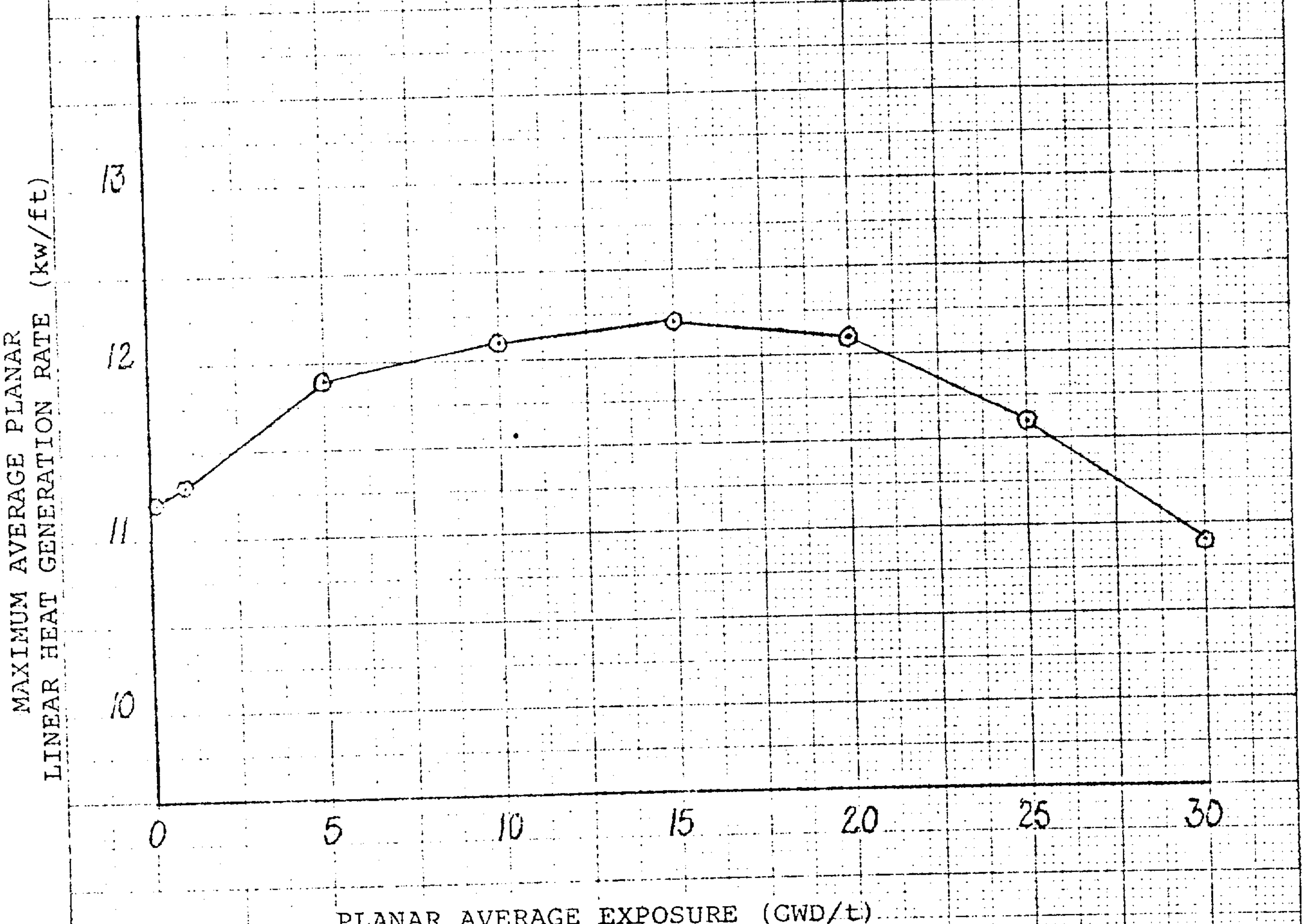


FIGURE 3.5-3 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

RELOAD 8D274L

REFERENCE
NEDO-21662-2
TABLE 5D

FULL CORE DRILLED

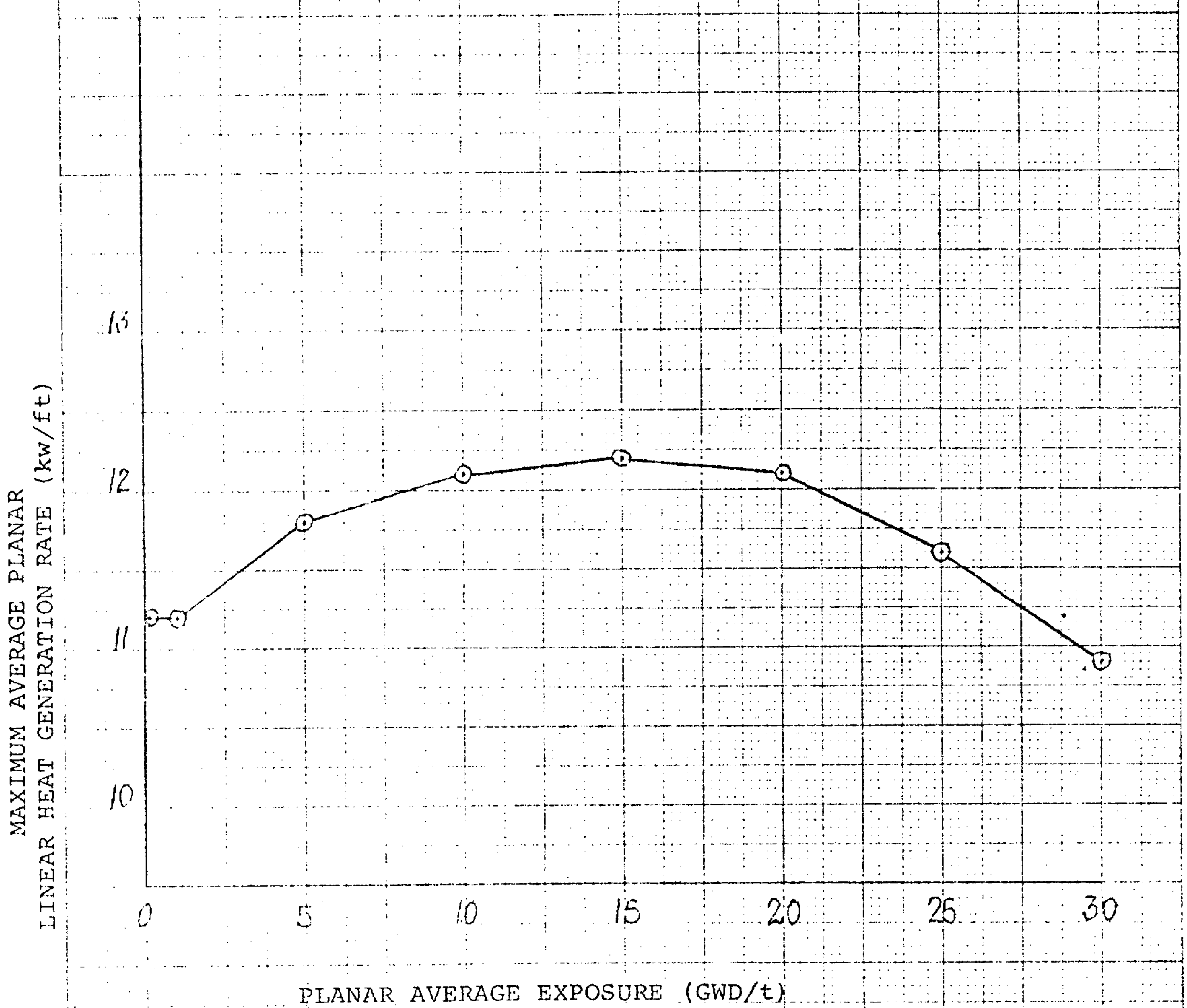


FIGURE 3.5-4 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

RELOAD 8D274H

FULL CORE DRILLED

REFERENCE
NEDO-21662-2
TABLE 5C

H. Jet Pump Flow Mismatch

1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122 percent the speed of the slower pump when core power is 80 percent or more of rated power, or 135 percent the speed of the slower pump when core power is below 80 percent of rated power.

2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.

3. The reactor shall not be operated for a total period in excess of 24 hours with one or more recirculation loops out of service.

1. The two recirculation loops have a flow imbalance of 15 percent or more when the pumps are operated at the same speed.

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10 percent.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the average of all jet pump differential pressures by more than 10 percent.

H. Jet Pump Flow Mismatch

1. Recirculation pump speeds shall be checked and logged at least once/day.

3.9 LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEMS

Applicability:

Applies to the auxiliary electrical systems.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Normal and Reserve A-C Power Systems

The reactor shall not be made critical unless all of the following requirements are satisfied:

1. Power is available to the emergency buses from the following power sources:

   a. the two 115 kv lines and reserve station service transformers

   b. the two Emergency Diesel Generator Systems.

2. a. 4,160 v buses 10,500 and 10,600 are energized.

   b. 600 v buses 11,500, 12,500, 11,600 and 12,600 are energized.

4.9 SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEMS

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Deleted

**F. LPCI MOV Independent Power Supplies**

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-165) are in service, except as specified below.

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 30 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason the following would apply.

   a. The other independent power supply including its charger, inverter, battery and associated bus is operable.

   b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

   c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its maintenance power source.

3. From and after the time that both independent power supplies are made or found inoperable continued reactor operation is permissible for 7 days, unless one of the trains is made operable sooner.

4.7 (Cont'd)

**F. LPCI MOV Independent Power Supplies**

1. Every week the specific gravity, voltage, and temperature of each pilot cell and overall battery voltage shall be measured and chargers and inverters shall be visually inspected.

2. Every three months the following measurements shall be made:

   a. Voltage of each cell to the nearest of 0.01V:

   b. Specific gravity of each cell;

   c. Temperature of every fifth cell.

3. Once every operating cycle the battery shall be subjected to a performance discharge test to verify battery capacity.

4. During power operation, if one of the LPCI MOV buses (MCC 155 or MCC 165) becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable bus is made operable sooner.

5. From and after the time both buses are made or found inoperable, the reactor shall be brought to cold condition within 24 hours.

6. **If conditions 2, 3, or 4 cannot be met, the reactor shall be brought to cold condition within 24 hours.**

NOTE: The above specifications 3.9.F.1, 2, 3, 4, 5, and 6 shall be in effect until the planned 1978 refueling outage after which time period the following 3.9.F.1, 2, and 3 shall become effective and will replace 3.9.F.1, 2, 3, 4, 5, and 6 above.

F. LPCI MOV Independent Power Supplies

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-165) are in service, except as specified below.

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason the following would apply.

   a. The other independent power supply including its charger, inverter, battery and associated bus is operable.

   b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

   c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its maintenance power source.

3. From and after the time both power supplies are made or found inoperable, the reactor shall be brought to cold condition within 24 hours.

## 3.9 BASES

The general objective of this specification is to assure an adequate source of electrical power to operate the auxiliary equipment during plant operation, to operate facilities to cool and lubricate the plant during shutdown, and to operate the engineered safeguards and Emergency Core Cooling Systems equipment following a loss-of-coolant accident. There are three sources of power available; namely, the normal a-c power source, the reserve a-c power source and the emergency a-c power source.

### A. Normal and Reserve A-C Power Systems

1. Normal plant a-c service power is supplied from a transformer connected to the main generator. This transformer is sized to carry 100 percent of plant auxiliary loads during normal operation. This transformer is not considered as a source of shutdown power since it is not available during shutdown conditions.

2. Reserve plant a-c service power is supplied from two transformers connected to the 115 Kv transmission system. Each of these transformers is sized to: (a) carry 50 percent of the plant auxiliary loads during station startup, and as a back-up supply for the normal source of a-c power; (b) to provide for maintenance and repair of equipment while retaining redundancy of power sources; and (c) as the primary source of a-c power for the engineered safeguards and Emergency Core Cooling Systems equipment.

   If one of the sources of reserve a-c power is not available the plant shall be permitted to run for 7 days provided that both emergency diesel generator systems are operable.

### B. Emergency A-C Power System

Emergency a-c power is supplied from two on-site redundant Emergency Diesel Generator Systems. Each system is designed to carry the redundant engineered safeguards loads for emergency core cooling required for safe shutdown of the plant and to maintain the plant in a safe shutdown condition following a loss of coolant accident with concurrent loss of normal and reserve a-c power sources.

C. Diesel Fuel

Day tank fuel oil capacity is based on operation of a pair of emergency diesel generator units operating at rated load for 3 hours. Minimum on-site fuel oil requirements are based on operation of one of the pairs of emergency diesel generators at rated load for 7 days. Storage tank capacities when maintained in full condition provide fuel oil capacity to permit operation of both Diesel Generator Systems for at least 7 days.

Additional diesel fuel can be delivered to the site within 48 hours.

If one of the Emergency Diesel Generator Systems is not operable, the plant shall be permitted to run for 7 days provided both sources of reserve power are operational. This is based on the following:

1. The operable Emergency Diesel Generator System is capable of carrying sufficient engineered safeguards and emergency core cooling system equipment to cover all loss-of-coolant accidents.

2. The reserve (offsite) power is highly reliable.

D. Battery System

125 v DC power is supplied from two plant batteries each sized to supply the required equipment at design power following a loss-of-coolant accident with a concurrent loss of normal and reserve power. Each battery is provided with a charger sized to maintain the battery in a fully charged state while supplying normal operating loads.

E. LPCI MOV Independent Power Supplies

There are two LPCI MOV Independent Power Supplies each consisting of a charger, rectifier, inverter and battery. Each independent power supply charger-rectifier is normally fed from the emergency A-C power supply system to maintain the battery in a fully charged state. In the event of a LOCA each independent power supply is automatically isolated from the Emergency A-C power system. The battery and inverter have sufficient capacity to power the MOV's essential to the operation of the LPCI System. A maintenance power source is provided for each LPCI MOV bus whereby in the event its independent power supply is out of service, the LPCI MOV bus may be energized directly from the Emergency A-C Power System.

## 4.9 BASES

The general objective of this specification is to check equipment operability, detect equipment failures and deterioration.

### A. Normal and Reserve A-C Power Systems

1. Reserve A-C Power Source

   The equipment is normally operated in the stand-by energized condition. Surveillance monitors are provided for determining its normal operability status both while in stand-by or during plant start-up and shutdown procedures. Insulation tests are conducted at specified intervals to determine the condition of insulation.

2. Auxiliary Equipment

   Mechanical and electrical tests are conducted at specified intervals to assure proper functioning of equipment.

### B. Emergency A-C Power System

The emergency Diesel Generator Systems are tested monthly to determine functional performance. Test procedures and intervals are specified to check for failure or deterioration in equipment and system operation since last use. Full load applied to the diesel unit is applied to prevent fouling of the engine; operation at equilibrium temperatures ensures there are no overheat problems.

During the monthly test, (a) the air starting systems are checked for automatic starting of the compressors and their ability to recharge the receivers, (b) the fuel oil transfer system is checked to ensure that the transfer pumps will refill the day tanks.

During the operating cycle test, a functional test of the emergency a-c power system is made by simulating a loss-of-coolant accident and a coincident loss of normal and reserve a-c power to the plant for checking proper operation of the system including sequencing of engineered safeguards and for Emergency Core Cooling System equipment.

### C. Diesel Fuel

Diesel fuel quality is checked at specified intervals to determine water content, micro-organism slime formation, etc., to ensure high reliability of engine operation.

D. Battery System

Measurements and electrical tests are conducted at specified intervals to provide indication of cell condition and to determine the discharge capability of the batteries.

E. LPCI MOV Independent Power Supply

Measurement and electrical tests are conducted at specified intervals to provide indication of cell condition, to determine the discharge capability of the battery.

## 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 10CFR 100.3.

### B.5.2 REACTOR

A. The initial core consists of not more than 560 fuel assemblies of 49 fuel rods each as described in Section 3.2 of the FSAR. After cycle 1 reloads will consist of 8x8 fuel assemblies of 63 fuel rods and one water rod each as described in NEDO-20360 (the standard General Electric submittal for 8x8 fuel) therefore after an appropriate number of reloads the core will consist of not more than 560 8x8 fuel assemblies of 63 fuel rods and one water rod each.

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.

5.5 (cont'd)

B. The Keff of the spent fuel storage pool is ≤0.90 under normal conditions, and <0.95 during abnormal conditions as described in Section 9.3 of the FSAR.

5.6 SEISMIC DESIGN

The reactor building and all engineered safeguards are designed on basis of dynamic analysis using acceleration response spectrum curves which are normalized to a ground motion of 0.80 g, for the Operating Basis Earthquake, and 0.15 g, for the Design Basis Earthquake.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333

## 1.0 Introduction

By letter dated May 16, 1977, as supplemented August 8 and August 25, 1977, and by letter dated July 25, 1977, as supplemented August 31, 1977, the Power Authority of the State of New York (PASNY or the licensee) requested amendments to Facility Operating License No. DPR-59. By letter dated July 7, 1977, as revised July 29, 1977, and supplemented August 25, 1977, the licensee submitted a reevaluation of the Emergency Core Cooling System performance in compliance with our Order for Modification of License dated March 11, 1977.

The amendments would modify the Technical Specifications for the James A. FitzPatrick Nuclear Power Plant (JAF or the facility) to: (1) permit operation of the facility with (a) 8X8 reload fuel bundles with channel liners of 100 mil wall thickness, (b) two bypass flow holes drilled in all reload fuel bundles and all initial core fuel remaining in the core after refueling, (c) all initial bypass holes in the core support plate plugged, and (d) limiting maximum average planar linear heat generation rates (MAPLHGR's) as determined by a reevaluation of the Emergency Core Cooling System (ECCS) performance, and (2) provide bases, surveillance requirements and limiting conditions for operation for the Low Pressure Coolant Injection System Motor Operated Valve Independent Power Supplies (LPCI-MOV-IPS), the Containment Atmosphere Dilution System and the Main Steam Leakage Collection System. Only the proposal involving the LPCI-MOV-IPS is being considered in this action. The LPCI-MOV-IPS are proposed to be installed during the current refueling outage, replacing the swing buses supplying power to the LPCI-MOV, which will be removed.

The licensee has also proposed by letter dated June 21, 1977, to operate the facility with the Control Rod Drive Hydraulic Return Line valve closed.

As a result of the licensee's proposal and our review, modification to the licensee's proposed Technical Specifications were necessary. These modifications were discussed with and agreed to by the licensee.

## 2.0 Evaluation

### 2.1 Nuclear Characteristics

The reload information presented in the licensing submittal[1] closely follows the guidelines of Appendix A of NEDO-20360[4]. Although NRC staff review of later supplements to this report is not complete, this topical report has been found tentatively acceptable for use in connection with BWR-4 reactors containing 8X8 reload fuel.

A total of 132 8X8 fuel bundles with an average U-235 enrichment of 2.74 wt% will be loaded throughout the core; 76 of the reload fuel bundles contain fuel rods having a high gadolinia content (8D274H) and 56 bundles contain rods having a low gadolinia content (8D274L). The core contains a total of 560 bundles. Thus, about 25% of the fuel bundles are being replaced for the reload.

The information in Reference 1 shows that the nuclear characteristics of the cycle 2 core, consisting of both the reload 8X8 fuel and the once burned 7X7 fuel, are very similar to the previous core. Typical nuclear characteristics of the reloaded core are given in Table 5-1 of Reference 1. The void coefficient of reactivity at a core average void content of 36.3 percent varies from $-12.1 \times 10^{-4}$ to $-11.6 \times 10^{-4}$ $\Delta k/K/\%V$. The Doppler coefficient, at a fuel temperature of 650°C, varies from $-1.225 \times 10^{-5}$ to $-1.120 \times 10^{-5}$ $\Delta k/K/°F$. Thus based on our review of the information presented in the JAF licensing submittal and the generic 8X8 reload topical report, it is concluded that fuel temperature and void dependent behavior of the reconstituted core will not differ significantly from that which has been previously reported for cycle 1 of the JAF reactor.

The cycle 2 minimum shutdown margin is 1.27%$\Delta$k. This meets the Technical Specification requirement that the core be at least 0.38% $\Delta$k subcritical in the most reactive operating state with the single most reactive control rod fully withdrawn and with all other rods fully inserted.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by at least 0.03 $\Delta K$ at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for JAF is that the effective multiplication factor, $K_{eff}$, of the fuel as stored in the fuel storage racks is equal to or less than 0.90 for normal storage conditions. This requirement is met if the uncontrolled infinite multiplication factor, $k_\infty$, of a fuel bundle in the reactor core configuration is less than or equal to 1.30. Reload fuel bundle types 8D274H and 8D274L at the peak reactivity point have a maximum $K_\infty$ of 1.216 and 1.238 respectively. Both fuel types, therefore, meet the Technical Specification fuel storage subcriticality requirement.

## 2.2 Mechanical Design

The two types of Reload 1 fuel assemblies have the same mechanical design and fuel bundle enrichments as the 8D274L and 8D274H fuel assemblies described in the 8X8 generic reload topical report(4), except for the channel wall thickness and the drilled bypass flow holes in the fuel bundle lower tie plate. The channel wall thickness for the reload fuel assemblies is nominally 0.100 inches whereas the standard product line fuel channels have a nominal 0.080 inch wall thickness.

Sufficient plenum volume has been provided above the fuel stack to assure that the increase in internal pressure caused by fission gas release, when combined with the other mechanical design basis loads, does not cause the stress intensity limits(4) to be exceeded.

The generic reload topical report (4) which is under review, has been found acceptable as a guide for use in connection with BWR reactors containing 8X8 reload fuel. The thicker (0.100 inch wall thickness) channels will result in greater margins for withstanding operating loads. On the basis of our review of the generic reload topical report and the reload submittal, we conclude that the Reload 1 fuel for the JAF reactor has an acceptable mechanical design.

2.3 Thermal-Hydraulics

The generic 8X8 reload topical report[4] and the General Electric Thermal Analysis Basis (GETAB)[6] are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of GETAB, based on the Minimum Critical Power Ratio (MCPR) concept, was used to establish the:

(1) fuel cladding integrity safety limit,

(2) limiting condition of operation such that the safety limit is not exceeded for normal operation and abnormal operational transients, and

(3) limiting conditions of operation such that the initial conditions assumed in the accident analyses are satisfied.

We have reviewed[7] the GETAB report and have found it acceptable for use in the above applications for 8X8 and 7X7 fuel assemblies.

The JAF cycle 2 thermal limits based on the GETAB report and the plant specific information provided by the licensee have been reviewed. Our evaluation of these limits is reported herein.

2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR is 1.06 for both 7X7 and 8X8 fuel types. With this safety limit, based on the GETAB statistical analysis, 99.9% of the fuel rods in the core are not expected to experience transition boiling for abnormal operational transients. The uncertainties in the core operating parameters, plant system operating parameters and the GEXL correlation (Reference 1, Table 4-1) when combined with the design relative bundle power histogram for the core, form the basis of the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for JAF during cycle 2 are the same or more conservative than those used in Table IV-1 of NEDO-10958[8].

The generic core selected for the GETAB statistical analysis is a typical 251 inch diameter vessel/764 fuel assemblies core. The generic GETAB statistical analysis results are conservative since the core bundle power histogram used for the GETAB application has more high power bundles than the most adverse bundle power distribution expected at any time during the second cycle of operation of JAF. This results in a conservative value of the safety limit MCPR which satisfies the 99.9% criterion.

We conclude that the proposed fuel integrity safety limit MCPR of 1.06 is acceptable for both the 7X7 and reload 8X8 fuel in the JAF reactor core during cycle 2.

### 2.3.2 Operating Limit MCPR

Various transient events will reduce the operating MCPR. To assure that the fuel cladding safety limit MCPR of 1.06 is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which results in the largest reduction in the critical power ratio (i.e., ΔMCPR). The licensee has submitted[1,2] the results of analyses of those transients which produce a significant decrease in MCPR. The types of anticipated abnormal operational transients evaluated were reactor pressure increase, feed-water temperature decrease, coolant flow increase, etc.

The most limiting abnormal operational transient from rated conditions in these categories for the 7X7 and 8X8 fuel was the turbine trip with failure of the bypass valves. The licensee analyzed this transient at exposures from beginning of cycle (BOC)-2 to 2000 MWd/t before end of cycle (EOC)-2 and 2000 MWd/t before EOC-2 to EOC-2 to determine the largest ΔMCPR for this transient during the cycle. The analysis was performed at burnups near and at the EOC-2 since the nuclear parameters tend to become more limiting toward the EOC. This approach encompasses those parameters which significantly affect the results of this limiting transient (i.e., void coefficient, Doppler coefficient, and scram reactivity function) that do not coincidently have their most limiting values at one burnup. The maximum ΔMCPR's for the 7X7 fuel and the 8X8 fuel which resulted from this transient analysis (assuming at least 104% of rated core power and 100% of rated core flow) occurred at the latter part of the cycle and were 0.22 and 0.30, respectively.

Addition of these ΔMCPR's to the safety limit MCPR (1.06) gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the maximum operating limit MCPR's are 1.28 for 7X7 fuel and 1.36 for 8X8 fuel, at rated core flow conditions.

The transient analyses include Design Conservatism Factors (DCF) of 0.80, 1.25, and 0.95 for the scram reactivity functions, void coefficient, and Doppler coefficient respectively. Until the generic review on the DCF's is complete, use of the above values in conjunction with other conservatisms, are considered acceptable. The initial MCPR's and initial conditions assumed in the transient analyses were equal to, or conservatively greater than, the established operating values. Thus, the combination of the above DCF's and other conservatisms used in the analyses provide conservative margins which are acceptable to the NRC staff.

The anticipated operational transient which causes the most severe reactor isolation is the turbine trip without bypass. Fast closure of the turbine stop valves therefore produce a large pressure increase in the reactor. The peak transient pressure is limited by opening of the safety/relief valves. The results of the transient analysis show that the peak steam line pressure is limited by the safety/relief valves to 1163 psig and the peak vessel pressure is 1210 psig at the bottom of the vessel. Therefore, the transient pressure is well below the ASME Pressure Vessel Code limit of 1375 psig. We find this acceptable.

A GE study(6) has shown that the required operating MCPR varies with the axial and local (pinwise) power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peaking was assumed to be representative of BOC conditions, located at the core midplane, with an axial peak-to-average of 1.40.

The bundle R-factors, which are a function of the local power peaking distribution assumed in the GETAB analysis, are also representative of a BOC condition. The R-factor values used were 1.098 for the 7X7 and 8X8 fuel. During the cycle the local peaking, and therefore, the R-factor, is reduced while the peak in the axial shape moves toward the bottom of the core. The amount by which the R-factor decreases from beginning to EOC would, by itself, increase the required operating limit MCPR by approximately one percent. This adverse effect on the MCPR is offset by a beneficial relocation of the axial peak to below the core midplane. Overall conservatism was applied in the determination of the required operating MCPR since the assumed axial and local peaking were representative of the BOC, which provides the most persistently adverse set of axial and local peaking conditions.

Therefore, for exposures from BOC-2 to 2000 MWd/t before EOC-2, the operational MCPR was determined to be 1.22 and 1.20 for the 7X7 and 8X8 fuel respectively. The most limiting MCPR's occurred for exposures from 2000 MWd/t before EOC-2 to the EOC-2. These limiting values of MCPR were determined to be 1.28 and 1.36 for the 7X7 and 8X8 fuel respectively. The above operating limit MCPR's, at rated flow, will assure that the fuel cladding integrity safety limit will not be exceeded during any anticipated abnormal operational transient during cycle 2 operations. Thus the above stated operating MCPR's are acceptable for the JAF reactor during cycle 2 operations.

### 2.3.3 Rod Withdrawal Error

The rod withdrawal error transient (RWE) is discussed in References 1 and 2 for worst case conditions. The event description and analysis assumptions for the RWE are given in Reference 4. These references indicate that the local power range monitors (LPRM's) will detect and alarm a high local power condition. However, if the reactor operator ignores the LPRM alarm, the rod block monitor (RBM) subsystem (set at 105% of full rated power at 100% core flow) will terminate the RWE transient in time to limit the maximum change in the critical power ratio to 0.16 for 7X7 fuel and 0.09 for 8X8 fuel. A RBM rod block occurring at 105% power and full core flow results in a peak linear heat generation rate of 19.44 kw/ft and 14.76 kw/ft for 7X7 and 8X8 fuels respectively. These calculated LHGR's are below the safety limit LHGR's for 7X7 and 8X8 fuels respectively and are acceptable.

The rod withdrawal error analysis is based on the most reactive reactor state and conservatively assumes no xenon, which maximizes the amount of excess reactivity inserted upon withdrawal of the maximum worth control rod from the core. The analysis also allows for the most severe rod block monitor detector failure configuration allowed by the Technical Specifications.

Comparing the RWE ΔMCPR for each fuel type with the ΔMCPR's for the turbine trip without bypass transient shows that the latter transient is limiting. Operating limit MCPR's, based on the previously discussed turbine trip without bypass transient, will preclude the localized RWE transient from exceeding the safety limit MCPR of 1.06. For this reason, the analysis performed for the RWE transient and the predicted consequences are acceptable.

2.3.4 Operating MCPR Limits for Less than Rated Flow

To assure that the safety limit MCPR is not violated for the limiting flow increase transient (recirculation pump speed control failure) starting from less than rated flow conditions, the licensee will operate JAF in conformance with the limiting conditions for operation as stated in paragraph 3.1.B of the Technical Specifications. This requires that for core flow rates less than full rated flow, the licensee shall maintain the MCPR above the minimum operating values.

The minimum MCPR values for less than full rated flow are equal to the MCPR for full rated flow multiplied by the respective $K_f$ factor values appearing in Figure 3.1.1 of the Technical Specifications. The $K_f$ factor curves were generically derived and assure that for the most limiting flow increase transients, occurring from less than rated core flow, the actual MCPR will not exceed the safety limit MCPR of 1.06.

Application of the above stated $K_f$ factors for reduced flow conditions results in calculated consequences for the limiting anticipated flow increase transients, which do not exceed the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

2.4 Accident Analysis

Our evaluation of postulated accidents affected by the actions being considered are discussed in the following sections.

2.4.1 ECCS Appendix K Analysis

In December of 1976 the NRC staff was informed that certain input errors and computer code errors had been made in the evaluations of ECCS performance for JAF. An Order was issued to the Power Authority of the State of New York on March 11, 1977(10), requiring that corrected "revised calculations fully conforming to the requirements of 10 CFR 50.46 are to be provided for the (James A. FitzPatrick) facility as soon as possible." Such corrected analyses were provided for the present reload in References 3 and 11. The corrected analyses included correction of all input errors previously made and correction of all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the NRC staff in a Safety Evaluation issued April 12, 1977(14).

We have reviewed the corrected analyses submitted for Reload 1 in Reference 3 along with supporting information submitted in Reference 11. We conclude that the JAF will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: 1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Tables 5A, 5B, 5C, and 5D of Reference 3; and 2) when it is operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the LOCA, as described elsewhere in this Safety Evaluation).

The analyses submitted in Reference 3, as supported by Reference 11, represent the first "lead plant" (i.e., complete break spectrum study) submitted with the corrected model. The analyses provide all information requested in the NRC letter to GE on June 30, 1977[12], regarding number of breaks to be analyzed, documentation to be provided, etc. for the new analyses. Since these analyses will be referenced by other BWR/4 plants with the LPCI system modification, the following description is provided of particular features of the analyses which must be considered by other plants referencing these analyses.

The break spectrum (i.e., peak clad temperature (PCT) vs. break size) shows that the particular break producing the highest PCT is a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break. That break for JAF is herein called the limiting break. Reasons why this break's analysis for JAF produces the highest PCT are presented below.

The limiting location is the recirculation pump discharge line rather than the larger diameter recirculation pump suction line due to the LPCI system modification previously made on JAF and other plants. The LPCI modification consisted of eliminating the loop-selection-logic system which previously had been provided to select the unbroken recirculation line following a LOCA and direct all LPCI flow from both LPCI systems to the unbroken recirculation line. (The loop-selection-logic system was subject to single failures, such as failure to open of the single LPCI discharge valve leading to the unbroken recirculation line. This failure would prevent all LPCI flow from both LPCI systems from entering the reactor). In place of the loop-selection-logic system, one LPCI system was permanently piped to one recirculation pump discharge line, and the other LPCI system was permanently piped to the second recirculation pump discharge line. After blowdown following a LOCA, the recirculation pump discharge valves close. These valves are located between the LPCI system injection point on the recirculation pump discharge line and any potential break location on the recirculation pump suction line. The LPCI system connected to the broken recirculation line is thus isolated from any suction line break (the other LPCI system is also isolated because of its connection to the unbroken line), and since only one LPCI loop can be disabled by any single failure, the largest (suction line) break can derive credit for earlier reflooding due to effectiveness of at least one LPCI system. This significantly reduces PCT calculated for the larger (suction line) break, and, for plants with the LPCI modifications, reduces it below PCT calculated for the smaller discharge line break. For the discharge line break, the LPCI system injection point cannot be isolated from the break location. Therefore, a break in a smaller diameter line than the suction line for plants without the LPCI modification would be expected to yield a lower PCT. For plants with the LPCI modifications, as with JAF, lack of LPCI flow* for the discharge line break delays the reflooding (with respect to the suction line break where LPCI flow from at least one system is available). This condition results in the discharge break for JAF being limiting. This result (discharge break limiting) has been observed previously and in fact was the reason behind design and implementation of the LPCI modification. (A MAPLHGR limit increase is realized by lowering of the previously limiting suction line break PCT.)

In summary, JAF represents the first instance in which a jet pump BWR has shown a limiting break size of less than 100% of the maximum possible limiting location break area. Three facts combine to cause these analysis results.

First, JAF is a 218" ID BWR/4 plant. These plants have a relatively large number of fuel assemblies in a relatively small core barrel (shroud) compared to other plants. Therefore, JAF's peripheral bypass area (the region between the shroud and the outer fuel channels) is small compared to other plants, making JAF more likely than other types of plants to experience Counter-Current-Flooding (CCFL) effects in the bypass area.

Second, in the time period since the last LOCA analysis was performed for JAF, holes have been drilled in the lower tie plates of all fuel bundles to enhance flow in the bypass area. These holes, at the bottom of the bypass region, are a major pathway for core spray water to reach the lower plenum following a LOCA and thereby contribute to the reflooding inventory, providing earlier reflooding and lower calculated PCT's. Any CCFL effects in the bypass area will delay such reflooding, causing a higher calculated PCT.

*One LPCI system cannot be isolated from the break and its flow is lost out the break; a single failure is assumed in the other system.

Third, a model change was made which results in a longer depressurization period and a lower minimum pressure for breaks smaller than the largest break for JAF. The change was made to insure that CCFL effects (penalties) are conservatively represented: at lower pressures, a given amount of steam produced will exit at a higher velocity since it occupies more volume; this will maximize CCFL effects. Lower plenum flashing is the primary contributor to steam flow in the bypass region. Lower plenum flashing occurs more abundantly and for a longer period for the new model's slower depressurization and lower pressures.

These three effects combine to make the 0.8 times the largest dischange break size limiting for JAF. The slower depressurization and lower pressure reached in the new model for the limiting break compared to the previously approved model's analysis of that break result in more lower plenum flashing over a longer time interval. The small 218" ID BWR/4 bypass region with these greater amounts of steam flow causes more CCFL effects to occur over a longer period of time in the bypass region than in analyses made with the previously approved model. Also, due to the newly drilled holes and the resulting greater importance of reflooding flow through the bypass region, more severe reflooding delays and PCT increases are experienced. These effects are most pronounced in the limiting break size region. For larger breaks (suction line and larger discharge line breaks), the depressurization is faster and lower plenum flashing is more near completion before the core spray flow is initiated (i.e., less steam is produced while spray flow is going down through the bypass area). For break sizes smaller than the limiting break, following blowdown a greater residual inventory is left in the lower plenum. This condition reduces the ECCS flow required to cause reflooding and more than compensates for any potential greater CCFL effects.

For JAF, the lead BWR/4 with LPCI modification, bounding calculation results have been provided for the limiting break and for one break slightly smaller and one break slightly larger than the limiting break. In addition, calculations for the largest suction line break, the largest discharge break and 1.0 $ft^2$ discharge line breaks have been provided. The bounding calculations showed PCT's slightly lower than the limiting break PCT.

Based on the foregoing, we conclude that this ECCS reevaluation fully meets the requirements of 10 CFR 50.46 and thereby satisfies the conditions of our Order for Modification of License dated March 11, 1977.

### 2.4.2 Steamline Break Accident

The spectrum of steamline break accidents which are postulated to occur inside containment are covered by the ECCS analysis. The analysis results and conclusions of steamline break accidents occurring outside containment, as presented by the licensee, are acceptable based on our generic review of NEDO-20360[4].

### 2.4.3 Fuel Loading Error

Fuel loading error is discussed in References 1 and 2 respectively for 8X8 fuel bundles placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in References 1 and 2 indicates that a fuel loading error results in a peak linear heat generation rate (LHGR) of 16.6 kw/ft in the misloaded 8X8 fuel bundle. The calculated peak LHGR is below that required to exceed the 1% plastic strain fuel design limit.

The present method used to calculate the MCPR of the worst case (rotated) fuel bundle indicated a MCPR of 1.00. Fuel bundles adjacent to the misloaded bundle are not affected, and the number of fuel rods in the misloaded bundle expected to experience transitional boiling is approximately four. Therefore, these fuel rods are presumed to fail by virture of mechanisms arising from the hostile transitional boiling regime.

A generic review of a more realistic method of calculating the MCPR of a misloaded bundle is now in progress. This new method will remove some of the extra conservatisms in the present method.

Until the new generic method is approved, the licensee was given the option for JAF of either: 1) increasing their Safety Limit MCPR by 0.06 or 2) providing Technical Specifications which would assure the detection of abnormal fuel degradations.

We have discussed the Technical Specification option with the licensee and the licensee has agreed to additional limits on the off-gas release rates. Therefore, detection of abnormal fuel degradation is accomplished by measurement of the radioactivity of the reactor coolant, measurements of the radiation levels in the main steam tunnel at the main steam isolation valves, and measurements of off-gas radioactivity rates at the air ejector.

As required by the Technical Specifications, samples of the coolant shall be analyzed for gross gamma activity prior to startup and at four hour intervals during startup. Thereafter samples shall be taken and analyzed at least every 96 hours. During steady state operations an increase in the off-gas at the air ejectors of .01 ci/sec within a 48 hour period or a power level change of $\geq$20% of full rate power/hour necessitates additional coolant sampling. The allowable limit for iodine in the reactor coolant, 3.1 u Ci/gm dose equivalent I-131, approximately corresponds to the levels expected immediately after gross failure of several pins. If the failure of a large number of fuel pins (in the order of 80) causes the off-gas activity to increase above 0.3 Ci/sec (30-minute decay value) for more than 15 minutes, an alarm will signal the operator to manually shutdown the reactor. Similarly, if the off-gas activity level increases above 1.0 ci/sec (30-minute decay value) for more than one minute, closure of the air ejector discharge valves would result in an automatic shut-down of the reactor.

This level would be exceeded under post-startup conditions if a few gross failures of fuel pins occurred sequentially and may be exceeded for a gross failure of a single pin in some cases. A third trip level setting resulting in reactor shutdown is closure of the main steam line isolation valves because of high radiation levels in the steam tunnel. This would occur at three times full normal power background radiation levels (caused mainly by short-lived N-16). These set points correspond to the levels that would result from failure of several fuel pins.

We conclude that the agreed to Technical Specifications for the JAF reactor provide assurance that significant abnormal fuel degradation, including that which might result from an undetected fuel loading error, would be detected and reported to the NRC and that reactor shutdown would automatically result in the event that large numbers of fuel pins experienced gross failure.

Any radioactivity which passed the main steamline isolation valves and air ejectors prior to their closures would be retained on the charcoal beds of the off-gas treatment system where it would decay to levels at which significant offsite exposures would not result. Even in the unlikely event that the activity collected on the charcoal beds were released by some unrelated independent event, the resultant offsite exposures would be well within the guidelines of 10 CFR Part 100.

In addition to the detection capabilities and Technical Specification requirements, the licensee's Quality Assurance procedures for verifying fuel position and independently and separately verifying that each fuel assembly was loaded into the correct position in its proper geometry was also considered.

Thus based on the above discussions, we find the licensee's approach to the fuel loading error acceptable.

### 2.4.4 Control Rod Drop Accident

The cycle 2 control rod drop accident for JAF is not entirely within the generic bounding analysis presented in Reference 4. The actual cycle 2 Doppler coefficient for the cold and hot startup conditions conservatively falls within the values assumed in the bounding analysis. The accident reactivity shape functions and the scram reactivity shape functions for both hot and cold startup conditions do not falls within the bounding analysis. Therefore, the licensee has performed a plant specific control rod drop accident for cycle 2. The resultant peak enthalpies from the specific analysis for the cold and hot startup cases were calculated to be less than the 280 cal/gm design limit.

The licensee also provided a comparative evaluation of the effects of transient xenon during restarts on the control rod worths. These results indicated control rod worths well below that required to achieve the 280 cal/gm design limit. Until the generic review on this subject is complete, the above comparative evaluation is considered acceptable since we concur that it indicates a factor of seven margin to the design limit.

Based on the above, we conclude that the group-notch withdrawal sequence can perform its intended function by keeping control rod reactivity worths within the safety design limits for the control rod drop accident at all operating conditions.

2.4.5 Fuel Handling Accident

The licensee states,[1] and we agree, that the fuel handling accident description, analysis and results provided in the Final Safety Analysis Report (FSAR) and discussed in the generic reload topical report[4] are applicable to the 8X8 reload fuel. That is, the total activity released to the environment and the resulting radiological exposures for the reload fuel will be less than those values presented in the FSAR for the 7X7 core. As identified in the FSAR, the radiological exposures for this accident with 7X7 fuel are well below the guidelines set forth in 10 CFR 100. Therefore, we conclude that the consequences of this accident for the 8X8 fuel will also be well below the 10 CFR 100 guidelines.

2.5 Overpressure Analysis

The licensee presented the results of an overpressure analysis to demonstrate that an adequate margin exists to the ASME code allowable vessel pressure, which is 110% of the vessel design pressure. The transient analyzed was the fast closure of all main steamline isolation valves with the conservative assumption that a reactor scram would occur on the second (high neutron flux) scram signal rather than the first (10% valve closure position switches). Two analyses were performed corresponding to 2000 MWd/t before EOC-2 and at EOC-2 at reactor power ratings of 104%. No credit was assumed for the relief function of the safety/relief valves and all safety valves were assumed operable. Results of these analyses show that the peak pressure at the bottom of the vessel would be limited to 1243 psig by self actuation (safety mode) of all the safety/relief valves. The licensee also referred to a study showing the sensitivity of peak vessel pressure to valve operability (Reference 9) as supplemental to the specific analysis for the JAF reload overpressure analysis. This sensitivity study shows that failure of one valve would result in a peak pressure at the bottom of the vessel of 1263 psig. Failure of one valve therefore results in a margin of 112 psig to the ASME code limit of 1375 psig (110% of 1250 psig). This result is acceptable to the NRC staff.

## 2.6 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in Reference 1. The results of the cycle 2 analysis show that the 7X7 and 8X8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is within the operational design guide in terms of decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core stability characteristics at both conditions are within the operational design guide. These results are acceptable to the NRC staff.

We have expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps. The concerns are motivated by increasing decay ratios in reload fuel cycles and improved fuel design.

Our concerns relate to both the consequences of operating at an ultimate decay ratio and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing our concerns through meetings, topical reports, and a test program.

A reactor core stability test program has been performed at Peach Bottom Unit No. 2 end of Cycle 2.

The test program is expected to be a significant aid in the resolution of our generic concerns on stability. The testing was performed during April 1977. The results from the testing will be provided to the NRC staff by the General Electric Company. The results will be used to refine the reactor stability analysis safety margins.

Until this issue has been resolved generically, the licensee will be required to restrict operations in the natural circulation flow mode. The licensee has agreed to this Technical Specification limitation. The restriction will provide a significant increase in the reactor core stability margins during cycle 2. On the basis of the foregoing, we consider the thermal-hydraulic stability to be acceptable.

### 2.6.1 Recirculation Pump Startup From the Natural Circulation Operational Mode

During recent BWR reload reviews, the question of recirculation pump startup from the natural circulation operational mode was raised. The pump startup could increase flow, collapse moderator voids, and subsequently result in a reactivity insertion transient. The consequences of such an accident sequence has not been previously evaluated, so that for this reload review, additional information was requested.

The licensee was requested to provide analyses and startup test results to show that the startup of recirculation pumps from natural circulation conditions does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed. An option was also afforded to preclude power operations, i.e., at >1% rated thermal power, in the natural circulation mode by Technical Specification. The licensee has agreed to incorporate Technical Specifications which preclude reactor operation in the natural circulation mode. We find this measure acceptable.

## 3.0 Physics Startup Testing

As part of our review of Reload 1 for JAF, the licensee was requested to provide a description of the cycle 2 physics startup test program. In response to that request, the physics startup test program was provided by the licensee in Reference 2. The physics startup tests, along with the tests required to assure compliance with the Technical Specifications, provide an acceptable physics startup test program.

## 4.0 LPCI-MOV Independent Power Supplies

In support of their proposed Technical Specifications for the LPCI-MOV-IPS, the licensee submitted by letter dated August 31, 1977, the final design details of the proposed LPCI-MOV-IPS. Prior to the current refueling outage, the emergency power sources that provided power to the LPCI-MOV's were the redundant diesel generator power sources through an arrangement of swing buses. Such an arrangement is one in which a bus is automatically transferred to one or the other of two redundant standby power sources. There is evidence based on operating experience and analytical considerations that such an arrangement renders the power sources vulnerable to common mode failures. Therefore, the licensee proposed to remove the swing buses and install two uninterruptible power supplies, one for each division of LPCI-MOV's. This would therefore, eliminate the possibility of the common mode failure of the main standby power sources. We have reviewed the licensee's design details and find that:

1. The proposed removal of the swing buses and installation of the LPCI-MOV-IPS would provide an increase in safety by the removal of a potential failure mode of the standby power sources.

2. The proposed power supplies would provide power sources to the LPCI-MOV that are independent of the existing onsite power source.

3. The proposed modification to the power supplies would bring the power supplies to the LPCI-MOV into conformance with Regulatory Guide 1.6.

4. The proposed power supplies have been designed with adequate capacity to handle the required transient and steady state loads.

5. Sufficient control, indication and monitors are proposed in the control room and locally at the power supply cabinets for the safe operation of the proposed power supplies.

Based upon our review and the above findings, we conclude that the proposed LPCI-MOV-IPS would provide an increase in safety and are acceptable for installation and operation and the proposed Technical Specifications for the LPCI-MOV-IPS are acceptable.

## 5.0 Operation with the Control Rod Drive (CRD) Return Line (RL) Valve Closed

In the last quarter of 1976, the licensee became aware of the possibility of cracks developing in the vicinity of the CRD RL nozzle of all BWR reactors. The cause of the cracking was determined to be the continued thermal cycling of the nozzle from the return line flow. As an interim measure to eliminate further crack development in the nozzle area, the licensee proposed by letter dated June 21, 1977, to close the CRD RL valve, thus stopping the flow through the nozzle. The interim solution was proposed for this refueling period because time did not permit the implementation of a permanent solution. As a permanent solution, the licensee proposed, during the next refueling outage, to reroute the RL such that the RL flow enters the primary coolant system at a point other than at the CRD RL nozzle.

To demonstrate the adequacy of the interim solution the licensee has made a series of adjustments and tests to verify that the CRD system will operate satisfactorily with the CRD RL valve closed.

If the CRD RL is valved off, the reverse flow of coolant through the CRD seals and then into the vessel would be slightly higher than with the valve open. Also, if the valve is closed, the reverse flow would include water from the drives themselves without being filtered. These conditions, we believe, would increase the likelihood of foreign material to collect in the drive mechanism over a period of time. This material could potentially adversely affect the operation of the CRD, and also cause accelerated wear. To compensate for this apparent reduced reliability of the CRD's, the licensee proposed to increase the surveillance requirements to approximately three times the current requirements. This proposed increase in surveillance would continue until the licensee could verify that the increased reversed flow through the CRD's would not affect the reliability of the CRD's, or until the RL would be rerouted.

While the interim solution would result in reduced water flowing into the reactor vessel due to the closure of the valves in the CRD RL, it would still be possible to line up the CRD pump so as to deliver rated flow to the reactor vessel as make up water should it become necessary.

Based on the above considerations, we conclude that the action would provide no decrease in margin of safety, the health and safety of the public will not be affected by the proposed actions, and therefore the proposed closing of the CRD RL valve and increased surveillance required by the proposed technical specifications are acceptable.

## 6.0 Conclusions

We conclude that the reevaluation of the ECCS performance submitted by the licensee meets the requirements of our Order for Modification of License dated March 11, 1977, and based on our evaluation of the applications and the available information and the requirements set forth above, it is acceptable for the licensee to proceed with cycle 2 operation in the manner proposed.

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

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

Dated: September 16, 1977

## REFERENCES


1. "General Electric Boiling Water Reactor Reload-1 License Amendment for the James A. FitzPatrick Nuclear Power Plant Full Core Drilled Conditions" NEDO-21619, Class 1, April 1977. Attachment A

2. Power Authority of the State of New York letter (George T. Berry) to U. S. Nuclear Regulatory Commission (Robert W. Reid) "James A. FitzPatrick Nuclear Power Plant - Responses to Reload 1 Questions," August 8, 1977.

3. Power Authority of the State of New York letter (George T. Berry) to U.S. Nuclear Regulatory Commission (Robert W. Reid), "Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant (Lead Plant)" NEDO-2166-2, Revision 2, July 29, 1977.

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

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

6. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," General Electric Company, BWR Systems Department, November 1973, NEDO-10958.

7. "Topical Report Evaluation of General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, September 1974.

8. General Electric letter (John A. Hinds) to U. S. Atomic Energy Commission (Walter Butler) "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports," NEDO-10958 and NEDE-10958, "General Electric BWR Thermal Analysis Basis (GETAB): , Data, Correlation and Design Application," July 1974.

9. General Electric letter (Ivan F. Stuart) to U. S. Nuclear Regulatory Commission (Victor Stello, Jr.). "Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressures to Valve Operability," December 23, 1975

10. U.S. Nuclear Regulatory Commission letter (Robert W. Reid) to Power Authority of the State of New York (George T. Berry) "James A. Fitz-Patrick Nuclear Power Plant-Order for Modification of License", March 11, 1977.

11. Supplemental Information to "Loss-of-Coolant Accident Analysis Report for James A. Fitzpatrick Nuclear Power Plant (Lead Plant)" NEDO-21662-2 Revision 2 August 1977.

12. U.S. Nuclear Regulatory Commission letter (Darrell G. Eisenhut) to General Electric Company (E.D. Fuller) "Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-Lead Plants, June 30, 1977.

13. Power Authority of the State of New York letter (George T. Berry) to Nuclear Regulatory Commission (R. W. Reid) "Additional Responses to Reload Questions, Docket No. 50-333," dated August 25, 1977.

14. NRC letter (K. Goller) to General Electric (G. Sherwood), LOCA Model Changes, April 12, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 30 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.

The amendment authorizes operation of the facility with: (1) 8x8 reload fuel bundles with 100 mil channels, (2) holes drilled in the lower tie plate of all reload fuel bundles and all first cycle fuel remaining in the core after refueling, (3) independent power supplies for the Low Pressure Coolant Injection System Motor Operated Valves, (4) the valve of the control rod drive hydraulic return line placed in the closed position and (5) limiting Maximum Average Planar Linear Heat Generation Rates as determined by a reevaluation of the Emergency Core Cooling System (ECCS) cooling performance. Effective upon issuance of this amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-59, is terminated.

The applications for the amendment comply 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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with Items 1 and 2 of this action was published in the FEDERAL REGISTER on June 23, 1977 (42 F.R. 31847). A similar Notice in connection with Item 5 of this action was published in the FEDERAL REGISTER on July 22, 1977 (42 F.R. 37608). No requests for a hearing or petition for leave to intervene were filed following these notices of the proposed action. Prior public notice of Items 3 and 4 was not required since they do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the applications for amendment submitted by letters dated May 16 and July 25, 1977, as supplemented, (2) the licensee's request dated July 7, 1977, as revised July 29, 1977, and supplemented, (3) Amendment No. 30 to License No. DPR-59, and (4) 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 E. Bridge Street, Oswego, New York.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16th day of September 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Gerald B. Zwetzig

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