REGULATORY DOCKET FILE COPY

JULY 1 1980

Docket No. 50-333

Mr. George T. Berry
President and Chief Operating
Officer
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your reload application submitted by letter dated March 4, 1980.

We have reviewed the General Electric submittal, NEDO-24242, "Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant, Reload 3," dated February 1980. The justification contained therein as well as the supporting information provided in Attachment B to your March 4, 1980 letter has been found acceptable for cycle 4, as limited by Section 2.2.2.2 of the enclosed Safety Evaluation Report.

In addition to approving the reload itself, the enclosed Technical Specifications define Senior Reactor Operator responsibilities during refueling. In 1974, licensees were requested to submit administrative control requirements. By letter dated July 6, 1979 you were requested to propose a change to your Technical Specifications to adopt the wording: "All Core Alterations shall be directly supervised by either a licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, who has no other concurrent responsibilities during this operation". By verbal discussion with members of your staff, we have reached agreement on the requirements of the enclosed specifications.

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

Sincerely,

Philip J. Polk

for Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures and ccs:
See next page

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FORM 318 (9-76) NRCM 0240 U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369

Mr. George T. Berry - 2 - JULY 1 1 1980

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

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CPB	CPB	RSB
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Amendment + Federal Register notice

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JULY 11 1980

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Although we have approved cycle 4 operation for FitzPatrick, please be advised that two areas of transient analysis methodology must be resolved prior to cycle 5; i.e., the GE GEXL correlation and the use of the REDY code. With respect to the GEXL correlation, the staff had previously concluded that the 8x8R GEXL correlation used by GE in the reload analysis for non-equilibrium cores has conservatisms which are equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the staff. However, the data supporting the application of GEXL to 8x8R retrofit fuel have never been submitted for staff review in accordance with established procedures. Therefore, we require that this data base be submitted so that the staff can complete its review and that this issue be resolved prior to operation in future cycles. Regarding the REDY code, for future cycles, this code will not be acceptable for use in calculating core response to pressurization transients. By letter dated January 23, 1980 from the NRC to the General Electric Company, we requested that that ODYN code we used for future transient analyses. Therefore, this code will be required for the reload 4 analyses.

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Operating Reactors Branch #2
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DATE	7/ /80	7/ /80	7/ /80	7/ /80	7/ /80	

cc:

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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. 49
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated March 4, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8007280 249

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

FOR THE NUCLEAR REGULATORY COMMISSION

for Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 11, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 49

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages.

Remove	Insert
6	6
15	15
18	18
20	20
29	29
30	30
31	31
35	35
43	43
58	58
72	72
73	73
95	95
96	96
102	102
103	103
108	108
123	123
124	124
125	125
130	130
	135e
	135f
245	245
247	247 and 247a

surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ± 25 percent. The interval as pertaining to instrument and electric surveillance shall never exceed one operating cycle. 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 assmelby which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power as calculated by application of the GEXL correlation (Reference NEDE-10958).

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 18.5 KW/ft for 7x7 bundles and 13.4 KW/ft for 8x8, 8x8R and P8x8R bundles.

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

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

V. Electrically Disarmed Control Rod

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

W. High Pressure Water Fire Protection System

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

X. Staggered Test Basis

A Staggered Test Basis shall consist of:

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

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

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 the notch 46 (∿ 4%) and notch 38 (∿ 21%) insertion.

The times for notch 24 (∿ 50%) and notch 04 (∿ 91%) 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.

2.1 BASES (cont'd)

c. APRM Flux Scram Trip Setting (Run Mode)(cont'd)

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

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

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

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

d. APRM Rod Block Trip Setting

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 provides an added level of protection before APRM Scram. 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 parallels that of the APRM Scram and provides margin to scram, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

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.

2.1 BASES (cont'd)

C. References

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

2. "General Electric Fuel Application" NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

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 D31.1 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-24242, Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant Reload 3, February 1980, shows that the main steam isolation valve closure transient, with flux scram, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. Figure 7 in NEDO-24242 presents the curve produced by this analysis. Reactor pressure is continuously indicated in the control room during operation.

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

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 50 msec.

B. Minimum Critical Power Ratio (MCPR)

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

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

B. Maximum Fraction of Limiting Power Density (MFLPD)

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

3.1 (Cont'd)

FUEL TYPE	MCPR OPERATING LIMIT FOR INCREMENTAL CYCLE 4 CORE AVERAGE EXPOSURE		
	BOC4 to 2GWd/t before EOC4	EOC4-2GWd/t to EOC4-1GWd/t	EOC4-1GWd/t to EOC4
	At RBM trip Level Setting S = 0.66W + 39%		
7x7	1.24	1.28	1.28
8x8	1.24	1.35	1.36
8x8R	1.24	1.35	1.36
P8x8R	1.24	1.37	1.38
	At RBM Trip Level Setting S = 0.66W + 40 or 41%		
7x7	1.27	1.28	1.28
8x8	1.24	1.35	1.36
8x8R	1.24	1.35	1.36
P8x8R	1.24	1.37	1.38
	At RBM Trip Level Setting S = 0.66W + 42%		
7x7	1.30	1.30	1.30
8x8	1.27	1.35	1.36
8x8R	1.25	1.35	1.36
P8x8R	1.25	1.37	1.38

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

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

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

3.1 BASES (cont'd)

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

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the start-up 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 4 (NEDO-24242) for various core exposures are given in Specification 3.1.B.

The ECCS performance analysis assumed reactor operation will be limited to MCPR, as described in NEDE-24011-P-A. 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 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont'd)

C. High Flux IRM

D. Scram Discharge Volume High Level

E. APRM 15% Power Trip

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

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

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

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

11. See Section 2.1.A.1.

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

where:

FRP = Fraction of rated thermal power (2436 MWt)

MFLPD = Maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 MW/ft for 8x8, 8x8R and P8x8R fuel.

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

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

S_n = Scram setting in percent of initial

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

3.2 BASES (cont'd)

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

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

The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence.

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 trips are set so that MCPR is maintained greater than the Safety Limit.

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 channgel 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)	$S \leq (0.66W+42\%)x \left[\frac{FRP}{MFLPD}\right]$	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	$S \leq 0.66W+K$ (8)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	$\geq$2% of full scale	8 Inst. Channels	(1)
3	IRM Detector not in Start-up Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$86.4% of full scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Start-up Position	(3)	4 Inst. Channels	(1)
2 (4)(5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)

NOTES FOR TABLE 3.2-3

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

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3 (cont'd)

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

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

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

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

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

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

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

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

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

   K = Intercept values of 39%, 40%, 41%, and 42% can be used with appropriate MCPR Limits from Section 3.1.B.

3.3 (cont'd)

6. During initial fuel loading or subsequent refueling, 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:

Control Rod Notch Position Observed	Average Scram Insertion Time (Sec)
46	0.338
38	0.923
24	1.992
04	3.554

6. Prior to control rod withdrawal for start-up 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 shall be scram time tested. During all scram time testing below 20% power the RWM shall be operable.

3.3 (cont'd)

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:

Control Rod Notch Position Observed	Average Scram Insertion Time (Sec)
46	0.361
38	0.977
24	2.112
04	3.764

4.3 (cont'd)

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.

3.3 and 4.3 BASES (cont'd)

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

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

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

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage.

This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

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

C. Scram Insertion Times

The Control Rod System is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. Scram insertion time and scram reactivity curves shown in NEDO-24242, Figures 2a, 2b and 2c were used in analyses of power transients to determine MCPR limits. The scram insertion time test criteria of Section 3.3.C.1 conform to the scram insertion times of NEDO-24242. Therefore, the required protection is provided.

3.3 and 4.3 BASES (cont'd)

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 preoperational 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. <u>Reactivity Anomalies</u>

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

3.4 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 subcritical condition considering the hot to cold reactivity swing, decay of xenon poisoning, uncertainties and biases in the analyses, 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 every 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 through 3.5.8. If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

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

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

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

H. Average Planar Linear Heat Generation Rate (APLHGR)

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

3.5 (cont'd)

I. Linear Heat Genration Rate (LHGR)

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

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

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

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

LT = Total core length = 12 feet

L = Axial position above bottom of core

G = 18.5 KW/ft for 7x7 fuel bundles
= 13.4 KW/ft for 8x8, 8x8R and P8x8R bundles

N = 0.026 for 7x7 fuel bundles
= 0.000 for 8x8, 8x8R and P8x8R fuel bundles

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

4.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

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

3.5 BASES

A. Core Spray System and Low Pressure Coolant injection (LPCI) Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

The loss-of-coolant analysis is referenced and described in General Electric Topical Report NEDE-24011-P-A.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems. No single failure of ECCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

Core spray distribution has been shown, in full scale tests of systems similar in design to that of the FitzPatrick Plant, to exceed the minimum requirements by at least 25 percent. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 113 psig.

The LPCI mode of the RHR System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI mode of

3.5 BASES (cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

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

H. Average Planar Linear Heat Generation Rate (APLHGR)

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

The peak cladding temperature folloing a postulated loss-of-coolant accident is primarly a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm 20^{\circ}F$ relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.5.1 through 3.5-8.

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.

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 distributuion. For LHGR to be a limiting value below 25% rated thermal power, the ratio of local LHGR to average LHGR would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

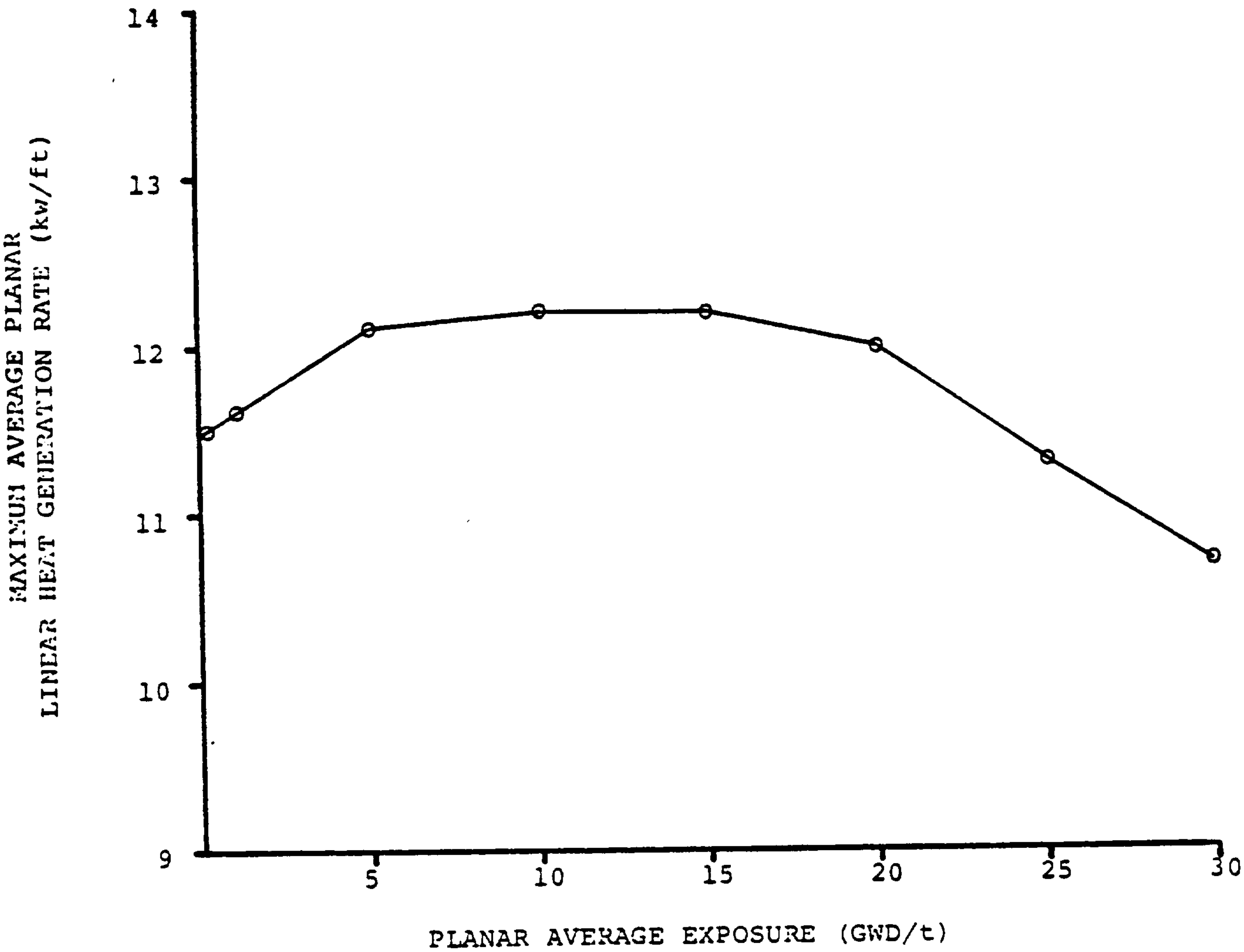

FIGURE 3.5-7 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

RELOAD 3, P8DRB265L

REFERENCE
NEDO-24242
SECTION 14

FULL CORE DRILLED

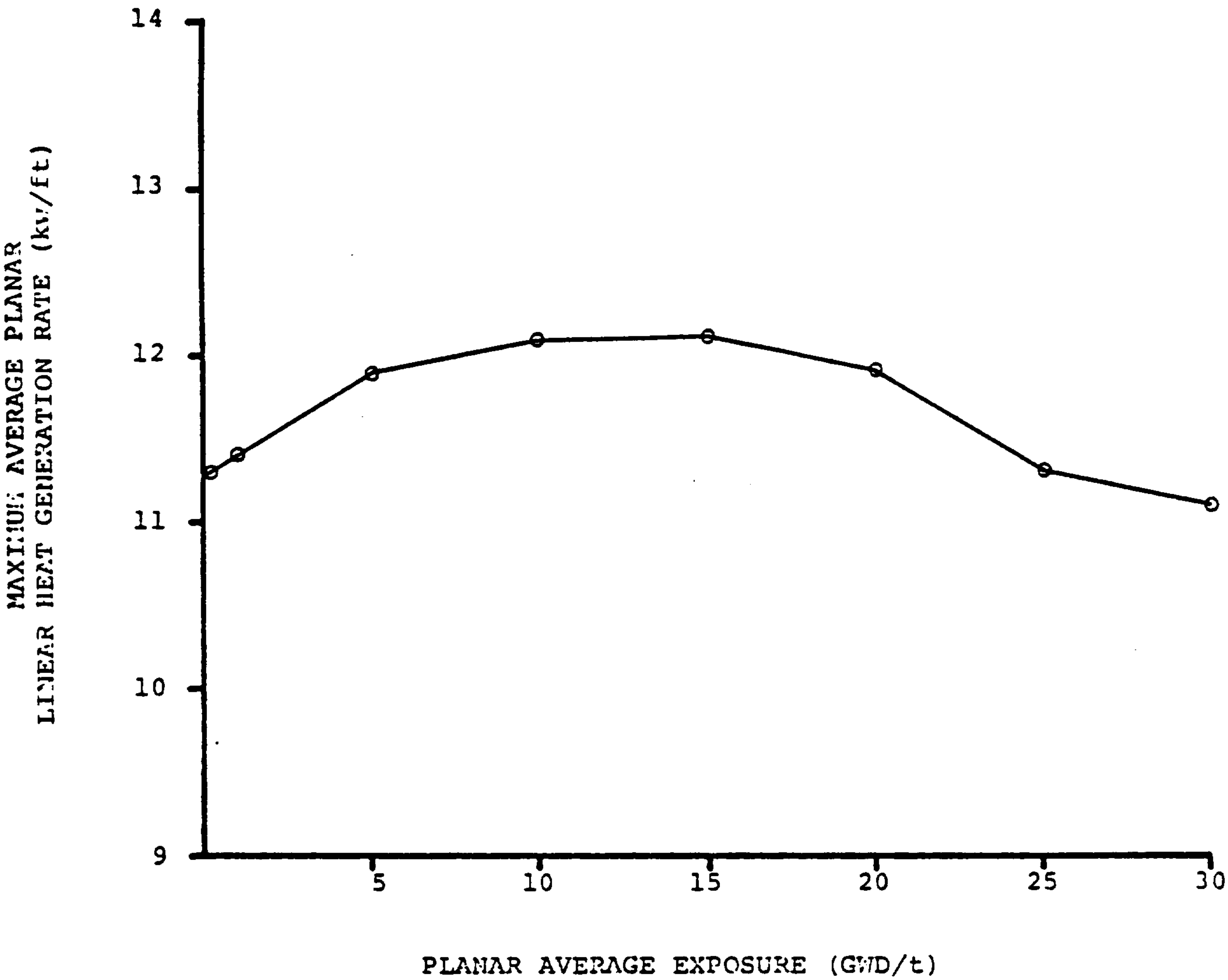

FIGURE 3.5-8 MAXIMUM AVERAGE PLANAR LINEAR HEAT RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

RELOAD 3, P8DRB283

FULL CORE DRILLED

REFERENCE
NEDO-24242
SECTION 14

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 areas as defined in 10 CFR 100.3.

5.2 REACTOR

A. The reactor core consists of not more than 560 fuel assemblies. For the current cycle four fuel types are present in the core: 7 x 7, 8 x 8, 8 x 8R and P8 x 8R. These fuel types are described in Section 3.2 of the FSAR and NEDO-24011. The 7 x 7 fuel has 49 fuel rods, the 8 x 8 fuel has 63 fuel rods and 1 water rod, and the 8 x 8R and P8 x 8R fuel have 62 fuel rods and 2 water rods.

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

5.3 REACTOR PRESSURE VESSEL

The reactor pressure vessel is as described in Table 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 of 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 K_{eff} dry is 0.90 and flooded is 0.95 described in Section 9.2 of the FSAR.

6.0 ADMINISTRATIVE CONTROLS

Administrative Controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of these measures are necessary to ensure safe and efficient facility operation.

6.1 RESPONSIBILITY

The Resident Manager is responsible for safe operation of the plant. During periods when the Resident Manager is unavailable, the Superintendent of Power will assume his responsibilities. In the event both are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel. The Resident Manager reports directly to the General Manager and Chief Engineer for administrative matters and functionally to the Manager - Nuclear Operations for operational related matters, as shown in Fig. 6.1-1.

6.2 PLANT STAFF ORGANIZATION

The plant staff organization is shown graphically in Fig. 6.2-1 and functions as follows:

1. A licensed senior reactor operator shall be on site at all times when there is fuel in the reactor.

2. In addition to item 1 above, a licensed reactor operator shall be in the control room at all times when there is fuel in the reactor.

3. In addition to items 1 & 2 above, a licensed reactor operator shall be readily available on site whenever the reactor is in other than cold condition.

4. Two licensed reactor operators shall be in the control room during startups and scheduled shutdowns.

5. A licensed senior reactor operator shall be responsible for all movement of new and irradiated fuel within the site boundary. A licensed reactor operator will be required to manipulate or directly supervise the manipulation of the controls of all fuel moving equipment, except the reactor building crane. All fuel movements by the reactor building crane, except new fuel movements from receipt through dry storage, shall be under the direct supervision of a licensed reactor operator. All fuel movements within the core shall be directly monitored by a member of the reactor analyst group.[a]

 SEE NEXT PAGE FOR FOOTNOTE

Footnotes:

(a) Paragraph 5 is effective until the end of cycle 4. During cycle 5 and thereafter, the following paragraph is effective:

5. A licensed senior reactor operator shall be responsible for all movement of new and irradiated fuel within the site boundary. All fuel movement as defined by Technical Specification section 1.B., "Core Alterations," shall be directly supervised by either a licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, who has no other concurrent responsibilities during this operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 49 TO LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

1.0 Introduction

By letter dated March 4, 1980,(Reference 1),the Power Authority of the State of New York has proposed changes to the Technical Specifications of the James A. FitzPatrick Nuclear Power Plant. The proposal documents the bases for the replacement of fuel assemblies for refueling of the core for cycle 4 operation and includes several other changes. The reload application included proposed Technical Specification changes in Reference 1 and was supported by the GE BWR supplemental licensing submittal (Reference 2).

This reload involves loading of prepressurized GE 8x8 retrofit (P8x8R) fuel. The description of the nuclear and mechanical designs of P8x8R fuel is contained in Reference 3. The use and safety implications of prepressurized fuel are presented in Reference 3 and have been found acceptable per Reference 4 (enclosed in Appendix C of Reference 3).

Values for plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other design parameters are provided in Reference 3. Additional plant and cycle dependent information is provided in the reload application (Reference 2) which closely follows the outline of Appendix A of Reference 3. Reference 4 includes a description of the staff's review, approval, and conditions of approval for the plant-specific data. The above-mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application in compliance with Reference 4.

Our safety evaluation of the GE generic reload licensing topical report has also concluded that the nuclear, and mechanical design of the 8x8R and P8x8R fuels, and GE's analytical methods as applied to mixed cores containing 7x7, 8x8, 8x8R and P8x8R fuels, are acceptable as limited by section 2.2.2.2. Approval of the application of the analytical methods did not include plants incorporating a prompt recirculation pump trip (RPT) or Thermal Power Monitor (TPM).

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Because of our review of a large number of generic considerations related to use of 8x8R and P8x8R fuels in mixed loadings, and on the basis of the evaluations which have been presented in Reference 3, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is directed to Reference 3.

2.0 Evaluation

2.1 Nuclear Characteristics

For cycle 4 operation, 24 new P8x8R fuel bundles of type P8DRB 265L and 136 new P8x8R fuel bundles of P8DRB 283 will be loaded into the core (Reference 2). The remainder of the 560 bundles in the core will be previously irradiated bundles as indicated in Reference 2. Based on the data provided in Reference 2 both the control rod system and the standby liquid control system will have acceptable shutdown capability during this cycle.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8x8 fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The 1.07 SLMCPR is incorporated into Technical Specifications. This is acceptable per Reference 3.

2.2.2 Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. Addition of the largest reductions in critical power ratio to the SLMCPR establishes the operating limits for each fuel type.

2.2.2.1 Transient Analysis Methods

The generic methods used for these calculations, including cycle-independent initial conditions and transient input parameters, are described in Reference 3. The staff evaluation, included as Appendix C of Reference 3, contains our acceptance of the cycle-independent values. Additionally, Appendix C contains our evaluation of the transient anlaysis methods, together with a description and summary of the outstanding issues associated with these methods. Supplementary cycle-independent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of Reference 2. Our evaluation of the methods used to develop these supplementary input values is also included in Appendix C of Reference 3.

2.2.2.2 Transient Analysis Results

The transients evaluated were the limiting pressurization and power increase transients (generator load rejection without bypass, inadvertent HPCI start, and feedwater controller failure and the control rod withdrawal error. The analysis results of the fuel loading error have been incorporated in the specification of the operating limit MCPR per Reference 3 (see Section 2.3.3). Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 2 were assumed.

The results of these analyses are outlined in Reference 2 sections 9 and 10. On this topic,it is acceptable if fuel specific operating limits are established for prepressurized fuel (Appendix C, Reference 3). On this basis, the transient analysis results are acceptable for use in the evaluation of the operating limit MCPR.

Although we have approved cycle 4 operation for FitzPatrick, please be advised that two areas of analysis methodology to predict the core response to transients are under generic review by the staff and could impact your MCPR operating limits in the near future. First, the staff has determined that the REDY code used for your transient analyses is in some instances non-conservative for evaluation of core response to anticipated transients; Consequently we will require that future analyses of the most limiting transients be performed with a code which provides acceptable best estimate calculation predictions. One such code is ODYN when applied in accordance with the licensing position described in our letter dated January 23, 1980 to the General Electric Company as augmented by subsequent written correspondence. The details of ODYN implementation for core reloads will be provided in the near future, and may involve recalculation of some limiting transients for cycle 4 in order to avoid a CPR margin penalty. Second, the test data base supporting the applicability of the GEXL critical power correlation to the retrofit (8x8R) fuel design has never been submitted for staff review in accordance with established procedures. Although we have approved operation of several reactors for up to two cycles with 8x8R fuel, we now have concerns regarding the safety limit MCPR predicted using GEXL for any fuel cycle with the two water rod fuel included in the core. Our concern relates to a possible non-conservative bias which has been observed in CPR test data for two water rod fuel with high pin-to-pin power peaking. We are now taking steps to resolve this concern via an expedited generic review. Until we have determined whether a non-conservatism exists, we will permit FitzPatrick to commence operation in the second cycle with the retrofit fuel.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

In our Safety Evaluation of Reference 3, we concluded that "the continued application of the present GE ECCS-LOCA ("Appendix K") models to the 8x8 retrofit reload fuel is generically acceptable and in our Reference 4 evaluation we extended that conclusion to prepressurized fuel. On this basis, the proposed MAPLHGR limits for the new prepressurized fuel are acceptable."

2.3.2 Control Rod Drop Accident

The significant parameters in the rod drop analysis satisfy the requirements for the bounding analyses described in Reference 3. Therefore, the results of this analysis are well below the acceptance criterion of 280 calories per gram.

2.3.3 Fuel Loading Error

The General Electric method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff and is part of the Reference 3 methodology. Potential fuel loading errors involving misoriented bundles and bundles loaded into incorrect positions have been analyzed by this methodology and the results have been incorporated into the specification for operating limit MCPR. This assures that SLMCPR is not violated for any potential fuel loading error.

2.3.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 3. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable.

2.4 Thermal Hydraulic Stability

The result of the thermal hydraulic stability analysis (Reference 3) show that the channel hydrodynamic and reactor core decay ratios at the natural circulation - 105% rod line intersection (which is the least stable physically attainable point of operation) are below the stability limit. Because operation in the natural circulation mode will be restricted by Technical Specifications, there will be added margin to the stability limit and this is acceptable.

2.5 Startup Test Program

The licensee has not changed his startup test program from that approved for the previous cycle. This program, therefore, remains acceptable.

2.6 Technical Specifications

The remaining Technical Specification changes are discussed in the following.

2.6.1 Administrative Changes

The majority of these Technical Specification changes are to reference the methods of Reference 3, General Electric's generic reload methodology and are administrative in nature.

The change in formulation from total peaking factor to a ratio of fraction of rated power and fraction of limiting power density to account for power peaking in the rod withdrawal block and flow biased APRM scram setpoints has been found acceptable. These two formulations are identical in their results but the proposed formulation eliminates the need for different peaking factors for different types of fuel. From a reactor protection viewpoint, this change is acceptable. However, in the bases the licensee has indicated that an adjustment in the APRM gain may be used to establish the peaking effect on setpoint. We have found this mode of calibration acceptable.

Because the new fuel has an increased active fuel length, the licensee has proposed a revised definition of top of active fuel which is reference to vessel zero and corresponds to the value used in the original fuel and FSAR. This is acceptable.

2.6.2 SRO Responsibilities

In 1974, the NRC requested that all power reactor licensees submit standard administrative control requirements. By subsequent letter dated July 6, 1979, the licensee was requested to comply with the prior NRC request (Reference 6). One of these requirements called for the direct supervision of core alterations by a licensed Senior Reactor Operator (SRO) who had no concurrent duties.

Section 6.2 of the James A. FitzPatrick Technical Specifications specifies that: (1) a licensed SRO shall be responsible for all movement of new and irradiated fuel within the site boundary; and (2) all fuel movements within the core shall be directly monitored by a member of the reactor analyst group. This does not satisfy the requirement of an SRO, without concurrent duties, supervising core alterations.

The NRC examination for a Senior Reactor Operator covers core alterations while the examination for a Reactor Operator does not. Therefore, a Senior Reactor Operator knowledgable in the affects of core alterations should direct refueling activities.

During a major outage, the currently required Senior Reactor Operator (assigned as the Shift Supervisor) can only devote a portion of time to any single activity because of the large number of activities for which he is responsible. For example, the administrative burden on the Senior Reactor Operator (Shift Supervisor) during a refueling outage includes several categories of work such as plant modifications, planned maintenance, preventive maintenance, annual overhaul, and in-service inspections. Many additional personnel are often assigned to the station during a major outage, which adds further burden for compliance with security requirements. Due to this, one Senior Reactor Operator cannot give refueling activities the attention they warrant. Therefore, a second, suitably qualified, person should be provided to direct those activities.

Accordingly, Section 6.2 of the Technical Specifications have been changed to adopt the wording: ALL CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, who has no other concurrent responsibilities during this operation.

3.0 Environmental Considerations

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

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a singificant decrease in a safety margin, the amendment does not involve a sigificant hazards consideration , (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 11, 1980

REFERENCES

1. Letter to Mr. Thomas A. Ippolito, (NRC) from Mr. Joseph R. Schmieder (PASNY) dated March 4, 1980.

2. "Supplemental Reload Licensing Submittal for James A. FitzPatrick Nuclear Power Plant, Reload 3," NEDO-24242, dated February, 1980.

3. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A, dated August, 1979.

4. Letter and enclosed SER to Mr. R. Gridley (General Electric) from Mr. Thomas A. Ippolito (NRC) dated April 16, 1979.

5. Memorandum to Mr. Thomas A. Ippolito (NRC) from Mr. Paul Check, "Review of Cooper Nuclear Station, Unit 1, Reload 4," dated April 11, 1979.

6. Letter to Mr. George T. Berry (PASNY) from Mr. Thomas A. Ippolito (NRC), dated July 6, 1979.

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. 49 to Facility Operating License No. DPR-59, issued to Power Authority of the State of New York, 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 is associated with the third refueling of FitzPatrick and changes the Technical Specifications to (1) include prepressurized 8x8 retrofit fuel, (2) revise operating limit minimum critical power ratios, and (3) incorporate administrative improvements.

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

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

For further details with respect to this action, see (1) the application for amendment dated March 4, 1980, (2) Amendment No. 49 to License No. DPR-59, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the State University College at Oswego, Penfield Library - Documents, Oswego, New York 13126. 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 Licensing.

Dated at Bethesda, Maryland this 11th day of July, 1980

FOR THE NUCLEAR REGULATORY COMMISSION

Philip J. Polk, Acting Chief
Operating Reactors Branch #2
Division of Licensing