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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 43 TO FACILITY LICENSE NO. DPR-59
POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333

## 1.0 Introduction

By letter dated August 18, 1978[1] and supplemented by letters dated October 13, 1978,[2] November 16, 1978[3] and November 17, 1978,[4] the Power Authority of the State of New York (PASNY), the licensee, requested amendment to the Technical Specifications appended to Operating License DPR-59 for James A. FitzPatrick Nuclear Power Plant (JAFNPP). The proposed changes relate to the refueling of JAFNPP, for Cycle 3 operation. It involves: (1) the replacement of 136 exposed 7x7 fuel assemblies with a like number of fresh, two water rod, retrofit 8x8 fuel assemblies (8x8R) designed and fabricated by the General Electric Company (GE); (2) the raising of setpoints and regrouping of reactor coolant system safety/relief valves (SRV) for Mark I Containment Short Term Program; and (3) modifications to the APRM rod block and trip setpoint formulation and system. In support of this reload application, the licensee has submitted a supplemental reload licensing document[5] prepared by GE, and proposed Technical Specification changes.[1-4]

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

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

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

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

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

## 2.0 Evaluation

### 2.1 Nuclear Characteristics

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

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

## 2.2 Thermal-Hydraulics

### 2.2.1 Fuel Cladding Integrity Safety Limit MCPR

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

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

### 2.2.2 Operating Limit MCPR

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

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

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

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

#### 2.2.3 Fuel Cladding Integrity Safety Limit LHGR

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

### 2.3 Accident Analysis

#### 2.3.1 ECCS Appendix K Analysis

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

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

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

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

2.3.2 Control Rod Drop Accident

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

2.3.3 Fuel Loading Error (FLE)

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

## 2.4 Overpressure Analysis

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

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

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

## 2.5 Thermal-Hydraulic Stability

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

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

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

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

## 3.0 Physics Startup Testing

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

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

## 4.0 Multiple-Consecutive Safety/Relief Valve Actuations

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

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

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

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

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

5.2 Modifications to the RPS for Thermal Power Monitor Installation

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

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

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

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

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

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

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

On these bases we find the proposed modification acceptable.

## 6.0 Environmental Consideration

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

## 7.0 Conclusion

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

Dated: November 22, 1978

References

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

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

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

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

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

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

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

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

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

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

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

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

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