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).

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.