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JPN-93-089

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Response to NRC Request for Additional Information
Power Uprate Submittal (TAC No. M83182)

REFERENCES:

1. NRC letter, Brian McCabe to R. E. Beedle, dated May 18, 1993, "Request for Additional Information - Power Uprate Submittal for the James A. FitzPatrick Nuclear Power Plant."
2. NYPA letter JPN-92-028, R. E. Beedle to NRC, dated June 5, 1992, "Proposed Changes to the Technical Specifications Regarding Power Uprate (JPTS-91-025)."

Dear Sir:

The Authority's response to the NRC questions (Reference 1) on the FitzPatrick power uprate program is included as Attachment I. The questions concern the structural analysis of the reactor vessel, internals, and various plant piping systems at power uprate conditions.

In addition, the Authority identified two errors in the initial power uprate submittal (Reference 2). The Technical Specification limits for Vessel High Pressure Scram were inadvertently listed in a table of analytical limits. The corrections are shown in Attachment II.

If you have any questions, please contact Mr. J. A. Gray, Jr.

Very truly yours,

W. A. Josiger Sor.

Ralph E. Beedle

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Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Question 1

(Section 2.5.1) - The evaluation did not address the effect of increased bottom head pressure on the structural and functional integrity of the control rod drive system (CRDS). Please state the basis for determining the acceptability of the CRDS regarding compliance with the Code. The information provided should include the Code edition, the Code allowables, the calculated maximum stresses, deformation, and fatigue usage factors for the uprated power conditions, and assumptions used in the calculations.

Answer 1

Section 2.5.1 of the power uprate safety analysis for FitzPatrick (Reference 1) discusses the performance of the Control Rod Drive Hydraulic System. Evaluations of other components in the Control Rod Drive System (CRDS) are summarized below. In addition, generic evaluations of the CRDS for uprate conditions are contained in the General Electric topical reports NEDC-31984P and NEDC-31897P-1 (References 2 and 3). These reports were provided to the NRC in both proprietary and non-proprietary versions.

CRDS Piping

The ASME code compliance evaluation of the CRDS piping for power uprate conditions is described in General Electric Report GE-NE-187-60-1191 (Reference 4). The results of this evaluation are reported in Section 3.5.1 of Reference 1. Temperatures and pressures for this evaluation were derived from the power uprate heat balance for the CRDS. The original design temperature and pressure bound the power uprate operating conditions so that new analyses are not required.

The code of record for the CRDS piping is ANSI B31.1 1967 Edition and Addenda through Winter 1969. The code does not require a fatigue analysis.

Design loads on CRDS pipe supports and at equipment interfaces also do not change since the design basis temperatures and pressures bound the power uprate operating conditions.

Control Rod Drive Mechanism

The nominal steady state operating pressure of the CRDS at power uprate conditions will be approximately 1075 psig. The design pressure of the CRDM is 1250 psig which exceeds the power uprate operating condition. Therefore, no further analysis of the CRDM is required.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Control Rod Drive Housing Nozzles

The Reactor Pressure Vessel Stress Report, updated for power uprate conditions (Reference 5) includes an evaluation of the Control Rod Drive Housing nozzles. These nozzles are the limiting component in the bottom head region of the reactor vessel. The code of record for the design and analysis of the Reactor Vessel is ASME Section III, 1965 Edition and Addenda through Winter 1966. The results of the evaluation are summarized in the response to Question 2 (Table 1).

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Question 2

(Sections 3.3.2 and 3.3.3) - The evaluation of the reactor vessel internals only considered the changes in reactor internal pressure. Please provide a discussion of how the dynamic effects of loss-of-coolant accident, jet reaction, and pipe restraint loads were accounted for in the evaluation of the reactor vessel and internals. The discussion should identity the Code and Edition used for evaluating stresses and allowables for the reactor vessel and internals. In addition, please indicate the maximum stresses, fatigue usage factor and location of the highest stressed area for both the current and uprated power conditions.

Answer 2

Consistent with the original FitzPatrick plant design basis, dynamic loads for LOCA were not considered in the evaluation of the reactor pressure vessel or reactor internals for power uprate. The reactor internal component evaluations did consider dynamic effects from seismic loads in the appropriate combinations with power uprate pressure differential loads (Reference 6).

The short term localized flow induced loads and acoustic loads which would follow a postulated recirculation line break were considered in the evaluation of the core shroud, shroud support, and jet pumps.

The original evaluation of the reactor pressure vessel considered the jet reaction loads from either a steam or recirculation line break. These analyses were not revised for power uprate, since the power uprate operating conditions are within the original design analysis values.

The pipe whip restraints for the Reactor Recirculation system were evaluated by General Electric. The power uprate operating conditions are within the original design basis temperature and pressure. Revised analyses are not required for reactor recirculation pipe whip restraints, calculated rupture restraint loads, or jet impingement effects, since the design basis High Energy Line Break (HELB) conditions envelope the power uprate operating conditions.

The code of record for the design and analysis of the reactor vessel is ASME Section III, 1965 Edition and Addenda through Winter 1966. The governing code of record for the feedwater nozzle is now ASME Section III, 1974 Edition and Addenda through Summer 1976, because of a design modification performed after initial fabrication.

The Reactor Pressure Vessel Stress Report was updated for the power uprate conditions (Reference 5). A summary of maximum stresses and fatigue usage factors for limiting components is provided in Table 1.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Table 1
SUMMARY OF POWER UPRATE STRESS AND FATIGUE RESULTS

Component	Original Fatigue Usage	Power Uprate Usage	Original P+Q SI (ksi)	Power Uprate P+Q SI (ksi)	Allowable P+Q SI (ksi)
Recirc. Inlet Nozzle	0.93	.931	64.4	66.6 34.8 (Note 1)	54
Feedwater Nozzle	0.804	0.962	65.0	67.3 15.5 (Note 1)	55.4
Control Rod Nozzle	0.780	0.852	35.2	36.5	69.9
Shroud Support	0.617	0.546 (Note 2)	104.4	106.8 29.1 (Note 1)	80.1
Vessel Shell	0.54	0.54 (Note 3)	40.2	40.2	80.1

NOTES

1. Excluding thermal bending stresses. The ASME Code (Section III, Subsection NB-3000) provides for the removal of thermal bending when the P+Q stress exceeds the allowable, provided that the following is also performed:

   a) Thermal ratcheting is shown to meet the criteria.
   b) A simplified elastic-plastic analysis is performed for the fatigue calculation.

   Thermal ratcheting was checked and no thermal ratcheting will occur. Also a simplified elastic-plastic analysis was performed for the fatigue calculations (Reference 5).

2. Decrease in fatigue usage factor is due to conservatism removed from original analysis.

3. The limiting stress cycles originally analyzed are not affected by Power Uprate conditions.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Question 3

(Section 3.5.2) - It appears that the adequacy evaluation for the reactor coolant pressure boundary piping, pipe supports, and equipment nozzles, has not been completed for the power uprate. Please provide a discussion regarding analysis methods, assumptions used, and compliance with the Code of record. The discussion should provide the Code allowables, the calculated maximum stresses, and fatigue usage factors for normal, upset and faulted conditions.

Answer 3

Pressure Boundary Piping Analysis

Evaluation of reactor coolant pressure boundary piping at power uprate conditions is complete. All pipe stresses were found to meet code allowables. The general methodology for performing pipe stress analysis is as follows:

1. Pipe stress analysis data (design pressure and temperature) are obtained for each affected system.

2. The corresponding uprate operating conditions are then compared to the design conditions for the affected system.

3. Systems whose original design conditions bound the uprated operating conditions are acceptable without further analysis. For example, the original stress analyses for the Feedwater and Main Steam piping inside containment were bounding for the uprate conditions. Table 2 shows the design pressure and temperature values used in the original analysis of these systems compared to uprate operating conditions.

4. Any system whose uprated conditions are beyond the original design analysis conditions, are evaluated using simple ratios for pressure and temperature. For example, an increase in thermal expansion stress is given by:

$$MAX\ S_{TH_{new}} = \frac{T_{new} - 70}{T_{old} - 70}\ (MAX\ S_{TH_{old}})$$

Similarly, an increase in longitudinal pressure stress is given by:

$$\Delta S_{LP} = \frac{\Delta P\, d^2}{D^2 - d^2}$$

Where: D = pipe outside diameter
d = pipe inside diameter

The new stresses are compared to allowable stresses. Material allowable stresses for pressure boundary piping are taken from the applicable code of record, ANSI B31.1 Piping Code 1967 through 1969 Winter Addenda. An example is provided below for the Reactor Water Cleanup System (line number: 4" - WR-902A-1).

Original analysis: T = 532 F P =1300 psig
Max. thermal stress: S_{TH} = 17789 psi

Uprate analysis: T = 536 F P = 1300 psig
New thermal stress: S_{TH} = 17941 psi

Code allowable = 22,500 psi (for thermal stress)

No fatigue analysis was performed for piping systems since this is not required by the original design basis or the code of record for the design.

Pipe Support Analysis

Evaluation of reactor coolant pressure boundary pipe supports is complete with the exception of fifteen supports. Nine supports are in the Residual Heat Removal System and six are in the High Pressure Coolant Injection System. One of the following three approaches is used to qualify pipe supports subjected to increased thermal loads at power uprate conditions:

1. The existing pipe support calculation is revised using a ratio of the existing input values and the new input values associated with power uprate conditions. This approach is sufficient for cases where the results of the revised calculation meet the established acceptance criteria.

2. The existing pipe stress analysis is re-evaluated to determine more exact loading values which can be used to refine the pipe support calculation and obtain acceptable results.

3. In the event that no analytical approach provides acceptable results, the existing support is redesigned and modified in the field.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Nozzle Analysis

Evaluation of reactor coolant pressure boundary equipment nozzles is complete with the exception of 2 nozzles in the High Pressure Coolant Injection System. Equipment nozzles affected by increases in pipe support thermal loads are evaluated in accordance with the FitzPatrick design basis document used for performing stress analyses (Reference 7).

The design and analysis of the equipment nozzles were based on ASME Boiler and Pressure Vessel Code, Section III. Depending on the purchase date of each piece of equipment, either the 1968 or 1971 edition was used. The 1968 edition provides requirements in terms of Class B and Class C vessels. The 1971 edition provides requirements in terms of Class 2 and Class 3 vessels. In general, Class B and Class C of the 1968 edition correspond to Class 2 and Class 3 of the 1971 edition, respectively. Requirements in both editions incorporate by reference certain design rules and requirements of ASME Boiler and Pressure Vessel Code, Section VIII.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Table 2

COMPARISON OF ORIGINAL DESIGN AND POWER UPRATE ANALYSIS CONDITIONS

System	Existing Analysis		Uprate Conditions	
	Design Pressure, psig	Design Temperature, F	Operating Pressure, psig	Operating Temperature, F
Main Steam piping in containment (System 29)	1250	583	1040	551
Feedwater piping in containment (System 34)	1850	575	1123 / 1067 (See Note 1)	369 / 423 (See Note 1)

Note 1: 1123 psig / 369 F are the nominal operating conditions at the discharge of the feedwater pump. 1067 psig / 423 F are the nominal operating conditions at the point where feedwater enters the reactor vessel.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Question 4

(Section 3.5.2) - This section stated that some supports may not meet design criteria at uprate and that three nozzles will be qualified with detailed analyses. Please provide more information regarding how these supports and nozzles were qualified for uprate.

Answer 4

There are three methods of qualifying the equipment nozzles. They are as follows:

1. Load Comparison
2. Qualification by vendor
3. Analytical method

If all three approaches fail to qualify the nozzle, modifications of the piping system will be performed to reduce the nozzle loads. This usually can be accomplished by relocating supports or rearranging piping.

The Authority will complete the evaluation and resolution of the remaining reactor coolant piping supports and nozzles prior to operating at power uprate conditions.

Question 5

(Sections 3.11.1 and 3.11.2) - Please indicate the code and edition used for the power uprate evaluation of balance of plant (BOP) piping and pipe supports including anchorages. It appears that the evaluation was not conclusive for all the BOP pipe supports and equipment nozzles. Please state the methodology, assumptions, and loading combinations used in the BOP pipe and pipe support analyses. In addition, please provide the code allowables, calculated maximum stresses and fatigue usage factors for normal, upset, and faulted conditions.

Answer 5

The analysis methods for the BOP piping, pipe supports (including anchorages) and nozzles are the same as those discussed in Answer 3 for the reactor coolant pressure boundary piping. The code of record for the design of BOP piping and components is ANSI B31.1.0 (1967), with Addenda A (1969) appended. Table 3 provides two examples of the BOP piping evaluations performed. A fatigue evaluation was not performed since it is not required by the code of record.

Evaluation of the BOP piping is complete with the exception of sixteen pipe supports and four nozzles in the Reactor Building Closed Loop Cooling Water System. The Authority will complete the evaluation and resolution of the remaining BOP piping supports and nozzles prior to operating at power uprate condition.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Table 3
SAMPLE BALANCE-OF-PLANT PIPING EVALUATIONS

System	Existing Analysis		Uprate Conditions	
	Design Pressure, psig	Design Temperature, F	Operating Pressure, psig	Operating Temperature, F
Feedwater Feed Pump Discharge Piping (System 34)	1850	375	1123 (See Note 1)	369
Feedwater Drain Piping From Heater 7A to Condenser (System 35)	4	135	2.3	137 (See Note 2)

NOTES

1. The uprate operating conditions are bounded by the existing analysis. There is no increase in stress loads.

2. This resulted in a maximum thermal expansion stress increase of 3.08%. The maximum thermal stress in this piping system is 2523 psi. The increase resulted in a maximum thermal stress of 2601 psi which is below the code allowable 22,500 psi.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

Question 6

(Section 3.11) - Please provide a discussion on how the current design basis analyses of pipe breaks, jet impingements, and qualification of safety related equipment are affected by the power uprate conditions.

Answer 6

The HELB (High Energy Line Break) engineering calculations that were performed as a requirement of NRC IE Bulletin 79-01 (Reference 8) were reviewed for the power uprate program. These calculations were performed using the methodologies and requirements found in NUREG-0588 (Reference 9). This engineering evaluation determined that the current design basis calculations envelope the power uprate operating conditions and thus are bounding. This is discussed in Chapter 10, page 10-1 of General Electric Report NEDC-32016P, "Power Uprate Analysis for the James A. FitzPatrick Nuclear Power Plant."

The design basis jet impingement loads calculated for the main steam and feedwater pipe tunnel and the suppression chamber are greater than loads which would result from power uprate operating pressure conditions (Reference 10). This is based on the design basis reactor dome pressure of 1050 psig being greater than the new operating pressure of 1040 psig.

The Authority's equipment qualification program was also reviewed as part of the power uprate program. The review included equipment in the drywell, reactor building, and steam tunnel for pressure, temperature, and radiation environmental parameters. The review concludes that existing equipment is qualified at power uprate conditions. Equipment qualification for power uprate conditions is discussed in detail in Chapter 10 of General Electric Report NEDC-32016P.

Response to NRC Request for Additional Information
FitzPatrick Power Uprate

References

1. GE Nuclear Energy Report NEDC-32016P, "Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant," dated December 1991.

2. GE Nuclear Energy Licensing Topical Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," dated July 1991.

3. GE Nuclear Energy Licensing Topical Report NEDC-31897P-1, "Generic Guidelines for Boiling Water Reactor Power Uprate."

4. GE Nuclear Energy Report GE-NE-187-60-1191, "Reactor Coolant Piping Systems Evaluation," dated November, 1991.

5. GE Nuclear Energy Report NEDC-32068, "Reactor Pressure Vessel Power Uprate Stress Report Reconciliation for the FitzPatrick Power Plant", dated March 1992.

6. GE Nuclear Energy letter BJB-9333 (Rev 1); C. Stoll to B. J. Branlund; dated October 5, 1993.

7. FitzPatrick Design Bases Document, "Design Criteria for Balance of Plant Piping Stress and Supports," Revision 0, dated April 1992.

8. NRC IE Bulletin 70-01, "Environmental Qualification of Class IE Equipment," dated February 8, 1979.

9. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," July 1981.

10. Stone & Webster Engineering Corporation, "Core Power Uprate Engineering Report for James A. FitzPatrick Nuclear Power Plant," December 1991.