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November 30, 1999
JPN-99-041

United States Nuclear Regulatory Commission
Attn: Document Control Desk
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Washington, D.C. 20555

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Discrepancies in NRC Safety Evaluation Report
Associated with Technical Specification (TS) Amendment 239

References: NRC Letter, K. Cotton to W.J. Cahill, Jr., Regarding Issuance of Amendment 239 for James A. FitzPatrick Nuclear Power Plant (TAC No. M92781), dated December 6, 1996

Dear Sir:

The Authority has identified several discrepancies in the NRC's Safety Evaluation Report (SER) associated with Technical Specification Amendment 239 (Reference) for FitzPatrick. The specific discrepancies are described in detail in Attachment 1 and have been categorized as either administrative, or typographical/grammatical. The Authority believes that these discrepancies are not safety significant.

The Authority requests that the NRC review the discrepancies listed in Attachment 1 and clarify the SER associated with Amendment 239, as appropriate.

There are no commitments made by the Authority in this letter. If you have any questions, please contact Ms. C. Faison.

Very truly yours,

Harry P. Salmon, Jr.
Vice President Engineering

cc: See next page

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Attachments:

1. Discrepancies in NRC Safety Evaluation Report Associated with Technical Specification (TS) Amendment 239

Attachment 1 to JPN-99-041

Discrepancies in NRC Safety Evaluation Report
Associated with Technical Specification (TS) Amendment 239

INTRODUCTION

The Authority has identified several discrepancies associated with the Technical Specification Amendment 239 (Power Uprate) NRC Safety Evaluation Report (Reference 1). These discrepancies have been classified as administrative, or typographical/grammatical. Specific discrepancies are detailed below.

I. ADMINISTRATIVE DISCREPANCIES

1. SER Section 2.0, page 2, states in part:

   "The operating pressure of the reactor will be increased approximately 25 psi to assure satisfactory turbine pressure control and pressure drop characteristics with the increased steam flow."

   The Power Uprate Safety Analysis Report (PUSAR) (Reference 2), page 1-5 states that reactor dome pressure is increased by 35 psi. Additionally, the value for reactor operating pressure in the hot standby condition in TS Section 1.0.D was changed from <1005 psig to <1040 psig.

   SER Section 2.0 should be revised - the reference to "25 psi" should be changed to "35 psi."

2. SER Section 3.1.4, page 3, addresses stability. The discussion on stability in the SER was no longer valid with the issuance of Amendment 236 (Reference 3). This amendment approved changes to establish operability requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with the Boiling Water Reactor Owners Group (BWROG) long-term solution Option I-D.

   SER Section 3.1.4 should be revised to reflect the information included in and conclusions of TS Amendment 236.

3. SER Section 3.1.5.1, page 4, states in part:

   "The CRD system was evaluated at the normal maximum reactor dome pressure of 1055 psig which is higher than the nominal power uprate operating pressure of 1040 psig for FitzPatrick."

   The PUSAR, page 2-3, states that the Control Rod Drive (CRD) system was evaluated at the uprated steam flow and dome pressure. PUSAR Table 1-2, page 1-11, states that the uprated power value for dome pressure is 1055 psia. In addition, Table 9-1 of the PUSAR states that the power uprate value for dome pressure used for the transient analysis was 1040 psig.

   SER Section 3.1.5.1 should be revised to state that the CRD system was evaluated at the uprated dome pressure of 1055 psia (1040 psig).

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4. SER Section 3.2.3, page 5, states in part:

   "Two SRVs were assumed to be out-of-service, and an initial operating pressure of 1055 psig was used in the analysis."

   PUSAR Table 9-1, "Parameters Used for Transient Analysis" shows that analysis dome pressure for power uprate was 1040 psig.

   SER Section 3.2.3 should be revised to reflect that 1055 psia (1040 psig) was used in this analysis.

5. SER Section 3.4.2.1, page 11, describes an analysis performed by the Authority for the Anticipated Transient Without Scram (ATWS) special event. This discussion on the ATWS special event in the SER was no longer valid with the issuance of TS Amendment 237 (Reference 4). This amendment approved changes to allow for a high-pressure trip setpoint change in the ATWS recirculation pump trip logic.

   SER Section 3.4.2.1 should be revised to reflect the information included in TS Amendment 237.

   In addition, the first sentence of SER Section 3.4.2.1, page 11, should be revised to refer to Reference 14 in the NRC SER (instead of Reference 13 in the NRC SER).

6. SER Section 3.5.1.1, page 14, first paragraph, states in part:

   "The licensee indicated that the required NPSH for RHR is 2.55 psig at the peak suppression pool temperature of 209F. Thus, adequate margin is available to prevent cavitation. The other ECCS pumps do not require credit for the increase in containment pressure due to the LOCA."

   The TS Bases, Section 3.7, page 188, approved for power uprate states in part:

   "For an initial maximum torus water temperature of 95F, assuming the worst case complement of containment cooling pumps (one LPCI pump and two RHR service water pumps), containment pressure is required to maintain adequate net positive suction head (NPSH) for the core spray and LPCI pumps."

   The power uprate safety evaluation (JPN-92-028, Reference 5), Attachment II, page 25 of 45, performed for the above TS bases change states in part:

   "The suppression chamber rises to 208.7F, as indicated in Table 4-1 of the PUSAR, when the single failure of the EDG is assumed. The pumps require up to 2 psig of torus pressure at this temperature...."

   NRC Letter (Reference 6), states in part:

   "Additionally, the staff concludes that the current use of containment

> *overpressure credit of less than 2 psig for the RHR and core spray pumps is consistent with the FitzPatrick licensing basis due to staff approval of power uprate and Amendment 239 of the FitzPatrick Technical Specifications. The staff notes that it would consider any future increase in reliance of containment overpressure credit, above 2 psig, for the RHR and/or core spray pumps to be an unreviewed safety question, which would require NRC approval to amend the operating license before the licensee could make such a change."*

Based on the above, Section 3.5.1.1 of the SER should be revised to reflect that reliance on up to two psig of containment pressure is acceptable for the purpose of maintaining adequate net positive suction head (NPSH) for the core spray and LPCI pumps.

7. SER Section 3.6.1, page 18, states in part:

> *"The primary and secondary air intake valves are manually operated from the control room to choose the most suitable air intake during the accident conditions."*

The Authority's submittal dated March 2, 1995 (Reference 7), page 10, Figure 1, shows that the primary and secondary manually operated air intake valves are located in FitzPatrick's HVAC (heating, ventilating and air conditioning) equipment room.

SER Section 3.6.1 should be revised to reflect the correct location of the primary and secondary air intake valves.

8. SER Section 3.6.1, page 19, states in part:

> *"For the DB main steam line break (MSLB), the licensee used an unfiltered inleakage of 100 cfm for the entire accident..."*

This statement is accurate for the Main Steam Line Break (MSLB) with spiked Reactor Coolant System (RCS) activity (Reference 8). However, for the design basis MSLB with equilibrium RCS concentration, unfiltered inleakage in excess of 100 cfm (i.e., 15,000 scfm pre-isolation and 2100 scfm post-isolation) was used in Reference 8.

SER Section 3.6.1 should be clarified to reflect that 100 cfm of inleakage was assumed for the MSLB under spiked RCS concentration conditions and that higher values were used in other analyses as detailed in Reference 8.

9. SER Section 3.8.1.2, page 24, third paragraph, third sentence, states in part:

> *"In its submittal, the licensee indicated that although the SW system return water temperature may increase, it will not exceed current Technical Specification (TS) limits and will have no effect on the capability and the design of the system."*

FitzPatrick does not have a TS regarding Service Water (SW) system return water temperature. The two sentences that precede the quote above accurately reflect what the Authority stated in PUSAR Section 6.4.1.2.

SER Section 3.8.1.2 should be revised to delete references to TS limits for service water system return temperature.

10. SER Section 3.8.5, page 25, last paragraph, last sentence, states in part:

> *"The licensee determined that the existing UHS system will continue to provide a sufficient quantity of water following a postulated LOCA for decay heat removal and that the TS for RHR reservoir level is adequate due to conservatism in the original water requirement calculations."*

FitzPatrick's TS do not include requirements for RHR (Residual Heat Removal) reservoir level, nor is RHR reservoir level part of the FitzPatrick design. As such, the Authority did not perform water requirement calculations for RHR reservoir level.

SER Section 3.8.5 should be revised to delete references to TS limits for RHR reservoir level. Information on the effects of power uprate on the FitzPatrick ultimate heat sink system is contained in PUSAR Section 6.4.5 "Ultimate Heat Sink."

11. SER Section 3.15.1, page 28, states there will be a slight increase in effluents as a result of power uprate. PUSAR Section 8.1 states that the quantities of liquid and solid radwaste will not increase. PUSAR Section 8.1 states the following:

> *"The liquid effluent doses reported in the UFSAR will increase based on the new GE radioactive source terms, but these doses would still be within the 10CFR20 and 10CFR50 Appendix I design objectives."*

SER Section 3.15.1 should be revised to distinguish changes in liquid effluent doses from changes in liquid effluent quantities.

12. SER Section 3.17.2, page 32, states in part:

> *"The licensee evaluated the adequacy of the control rod drive mechanism (CRDM) in accordance with the code of record, the ASME Code Section III, 1965 Edition and Addenda through Winter 1966."*

The FitzPatrick UFSAR, Section 3.5.5.1 states in part:

> *"Pressure containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the 1965 ASME Boiler and Pressure Vessel Code with the Winter 1966 Addenda.*
>
> *In 1986, approximately 10 Control Rod Drives were replaced by those of an*

> *improved design... Changes included a material substitution, minor fabrication changes, and larger cooling water orifice size.*
>
> *By May 1992, 45% of the originally installed control rod drive (CRD), had been replaced with those of an improved design of the BWR/6 Type. The design features include: 1) improved material for the cylinder tube and flange, and index and piston tubes 2) an improved cooling water orifice 3) an improved uncoupling rod, 4) an improved buffer design."*

Several CRDMs were modified, as described above, and comply with the stress requirements of ASME III 1971 Edition, with the Winter 1972 Addenda, and including the 1974 Edition with Winter 1975 Addenda. Under ASME Section XI, utilizing the original or later version of the ASME Section III code is allowed.

The Authority informed the NRC in Reference 9, (Attachment 1, Answer to Question 1), that the Control Rod Drive housing nozzles were evaluated in accordance with the code of record, the ASME Code Section III, 1965 Edition and Addenda through Winter 1966. Note that the housing nozzles were evaluated, not the entire CRDM.

SER Section 3.17.2 should be clarified to reflect the different ASME editions and addenda used by the Authority in its evaluation of the FitzPatrick control rod drive system.

13. SER Section 3.18, Item 1, states to change the allowable value for Reactor Vessel Steam Dome Pressure High on TS Table 3.1-1 from 1120 psig to 1195 psig. No allowable value is included on TS Table 3.1-1. The Authority did not propose this change in the power uprate submittal. The only change from 1120 psig to 1195 psig regards RCIC and HPCI testing in Section 4.5 of the TS.

    SER Section 3.18, Item 1, should be revised to delete the following statement:

    > *"Change Allowable Value from 1120 psig to 1195 psig"*

14. SER Section 3.20, page 40, second paragraph, last sentence states the following, in part:

    > *"The licensee reported the following: the percentage of core damage frequency for the first six actions was less than 0.1 percent, the percentage of core damage frequency change for the seventh operator action was 0.65 percent, and the total percentage of core damage frequency change was 0.65 percent."*

    The Authority stated in Reference 10, (Attachment I, page 16 of 27), that the total percentage of core damage frequency change was less than 1.0 percent.

    SER Section 3.20 should be updated to reflect a total core damage frequency change of less than 1.0 percent.

15. SER Section 3.1.3, "Power/Flow Operating Map," page 3, reflects language provided in Revision 0 to the Power Uprate Safety Analysis Report (PUSAR) (Reference 11). This information was updated with Revision 1 to the PUSAR (Reference 2).

    Specifically, the power/flow map is plotted in percent of rated both for flow and power. The rated power level does not appear on the map. The first sentence in Section 3.1.3 states, in part, power uprate is achieved by expanding the upper portion of the operating map (power/core flow) along the current rod/flow control lines. This is not the case. The same power/flow coordinates are used for pre and post power uprate analyses. Therefore, there is no expansion of the map.

    The first sentence further states that rod/flow control lines have been renamed to reflect the definition of rated thermal power. This is not the case. The rated rod/flow control line is expressed in percent of rated not megawatts.

    The second sentence in Section 3.1.3 discusses changes to the limits on the map due to power uprate. These changes are not consistent with the revised PUSAR. There is no change to the allowable flow range at full power under power uprate conditions.

    SER Section 3.1.3 should be revised to reflect Revision 1 to the PUSAR as detailed above.

16. SER Section 3.2.3, page 5, states in part:

    > *"The results of the analysis are documented in GE Topical Report NEDC-32016P."*

    The results of the analysis are documented in NEDC-32016P-1 (Reference 2, PUSAR). This report superseded NEDC-32016P (Reference 11) and was transmitted to the NRC under the Reference 12 letter.

    SER Section 3.2.3 should be revised to reference NEDC-32016P-1 and delete reference to superseded versions.

17. SER Section 3.2.4, page 6, references a letter sent by the Authority to the NRC dated November 14, 1996 (JPN-96-043, Reference 13). Due to an administrative error, the wrong attachments were included in this letter. Based on this, the Authority forwarded correct attachments to the NRC by letter dated November 20, 1996 (JPN-96-045, Reference 10). The revised attachments supersede those sent in the letter dated November 14, 1996.

    SER Section 3.2.4 should be revised to reflect the Authority's November 20, 1996 letter.

18. SER Section 3.2.6, page 6, states in part:

"The Licensee has assessed the RCIC system in a manner consistent with the bases and conclusions of Section 4.2 of Reference 1."

Reference 1 was a letter sent by the Authority to the NRC dated June 12, 1992 (JPN-92-028, Reference 5). This letter does not have a Section 4.2.

Section 3.8 of the PUSAR states that the Reactor Core Isolation Cooling (RCIC) system has been assessed to be consistent with the bases and conclusions of the generic evaluation in Section 4.2 of Reference 1. This PUSAR reference is NEDC-31984P and NEDO-31984 (Reference 14) and is identified in the NRC SER as Reference 4.

SER Section 3.2.6 should be revised to refer to Reference 4 of the NRC SER.

19. SER Section 3.2.7, page 7, states in part:

    "The LPCI mode is discussed in Section 4.2.2 of this report."

    Section 4.2.2 does not exist in the NRC SER. The low pressure coolant injection (LPCI) mode is discussed in Section 4.2.2 of the PUSAR, which is Reference 5 in the NRC SER.

    SER Section 3.2.7 should be revised to state that the LPCI mode is discussed in Section 4.2.2 of Reference 5.

20. SER Section 3.6.1, page 20, third paragraph, states that the staff has re-reviewed Section 4.5.4.

    SER Section 3.6.1 should be revised to clarify that the NRC staff re-reviewed Section 4.5.4 of Reference 5.

21. SER Section 3.17.1, page 31, second paragraph, last sentence, states in part:

    "The licensee recalculated RIPDs for the power uprate shown in Table 3-1 of Reference 6, for normal, upset and faulted conditions."

    Reference 6 of the NRC SER identifies the SAFER/GESTR-LOCA analysis for FitzPatrick. The Authority re-calculated Reactor Internal Pressure Differences (RIPDs) for the power uprate, which were shown in Table 3-1 of the PUSAR. The PUSAR was Reference 5 in the NRC SER.

    SER Section 3.17.1 should be revised to refer to Reference 5 (instead of Reference 6).

22. SER Section 3.20, page 42, third paragraph, second sentence, should be revised to reference letter dated November 20, 1996, JPN-96-045 (instead of the letter dated November 14, 1996, JPN-96-043).

SER Section 3.20, page 42, third paragraph, second sentence, the fifth bullet under this paragraph should be revised to refer to Question 16 (instead of Question 17).

23. SER Section 3.25, page 45, contains the following discrepancies:

a. The second paragraph (Reactor Parameters) refers to TS page 134 as a proposed change. This page was deleted under Amendment 236. No change to this page was required for power uprate.

SER Section 3.25 should be revised to delete the reference to TS page 134.

b. The fourth paragraph (Setpoints) refers to TS pages 41a and 42 as proposed changes and the seventh paragraph (Administrative) refers to TS page 41a as a proposed change. The Authority informed the NRC (JAFP-96-0306, Reference 15) that changes previously located on TS pages 41a and 42 were now located on TS pages 41 and 43 due to interposing amendments.

SER Section 3.25 should be revised to refer to TS pages 41 and 43 (instead of pages 41a and 42).

c. The sixth paragraph (Testing) refers to TS page 172 as a proposed change. The Authority informed the NRC (JPN-96-046, Reference 16) that Amendment 234 deleted TS page 172 and relocated TS Section 4.7.A.2.d.(1) to TS page 166 and renumbered this section as 4.7.A.2.c. Based on this, TS page 166 contains the change previously located on page 172.

SER Section 3.25 should be revised to refer to TS page 166 (instead of page 172).

24. SER Section 3.25, page 48, under changes to TS page 20, references report NEDC-32018P-1. Refer to Reference 5 in the NRC SER and TS Bases Section 2.1.C.1.

SER Section 3.25, page 48, should be revised to reference NEDC-32016P-1 (instead of NEDC-32018P-1).

Discrepancies in NRC Safety Evaluation Report Associated with Technical Specification (TS) Amendment 239

II. GRAMMATICAL/TYPOGRAPHICAL

1. SER Section 3.3.2, page 10, first paragraph, seventh sentence, third word, should be revised to replace "that" with "than."

2. SER Section 3.5.2.1, page 15, last paragraph, eighth sentence, should be revised to replace the acronym "LIP" with "LTP."

3. SER Section 3.15.3, page 29, first paragraph, fourth sentence is a duplicate of the third sentence and should be deleted.

4. SER Section 3.25, page 57, under changes to TS page 254c, Item Number 4, should be revised to replace the word "Supplemental" with "Supplement."

Attachment 1 to JPN-99-041

Discrepancies in NRC Safety Evaluation Report
Associated with Technical Specification (TS) Amendment 239

REFERENCES

1. NRC Letter, K. Cotton to W.J. Cahill, Jr., Regarding Issuance of Amendment 239 for James A. FitzPatrick Nuclear Power Plant (TAC No. M92781), dated December 6, 1996

2. General Electric Report, NEDC-32016P-1, Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant, Revision 1, Class III, dated April 1993.

3. NRC Letter to W.J. Cahill, Jr., Regarding Issuance of Amendment 236 for James A. FitzPatrick Nuclear Power Plant (TAC No. M95097), dated October 30, 1996.

4. NRC Letter to W. J. Cahill, Jr., Regarding Issuance of Amendment 237 for James A. FitzPatrick Nuclear Power Plant (TAC NO. M95523), dated November 7, 1996.

5. NYPA Letter, R. E. Beedle to the NRC, Proposed changes to the Technical Specifications Regarding Power Uprate (JPN-92-028), dated June 12, 1992.

6. NRC Letter to J. Knubel, Regarding NRC Bulletin 96-03 (TAC No. M96146), dated December 4, 1998.

7. NYPA Letter, W. J. Cahill, Jr. to the NRC, Response to NUREG-0737, Item III.D.3.4 Control Room Habitability (JPN-95-010), dated March 2, 1995.

8. James A. FitzPatrick Nuclear Power Plant, Updated Final Safety Analysis Report (UFSAR), Section 14.8.2, Uprate Power Level Radiological Analysis, revised May, 1997.

9. NYPA Letter, R. E. Beedle to the NRC, "Response to Request for Additional Information Power Uprate Submittal (TAC No. M83182)," (JPN-93-089), dated December 29, 1993.

10. NYPA Letter, W. J. Cahill, Jr. to the NRC, Incorrect Attachments Included with Response to Request for Additional Information on Power Uprate (JPN-96-045), dated November 20, 1996.

11. General Electric Report, NEDC-32016P, Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant, Class III, dated December 1991.

12. NYPA Letter, R.E. Beedle to the NRC, Proposed Changes to the Technical Specifications Regarding Power Uprate (JPTS-91-025) Revision 1 to Safety Analysis NEDC-32016P (JPN-93-060), dated August 18, 1993.

13. NYPA Letter, W. J. Cahill, Jr. to the NRC, Response to Request for Additional Information Regarding Power Uprate (JPN-96-043), dated November 14, 1996.

14. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor

Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992 and Supplements 1 and 2.

15. NYPA Letter, M. J Colomb to the NRC, Updated Page Changes for Proposed Change to the Technical Specifications Regarding Power Uprate (JAFP-96-0306), dated August 15, 1996.

16. NYPA Letter, W. J. Cahill, Jr. to the NRC, Supplemental Information Regarding Updated Page Changes for Proposed Change to the Technical Specifications Regarding Power Uprate (JPN-96-046), dated November 20, 1996.