UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 21, 2016

Site Vice President
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - REQUEST FOR ADDITIONAL INFORMATION REGARDING: LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS SECTION 5.5.6 FOR EXTENSION OF TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES (CAC NO. MF8305)

Dear Sir or Madam:

By application dated August 29, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16242A332), Entergy Nuclear Operations, Inc. (Entergy), submitted an application for a proposed amendment for James A. Fitzpatrick Nuclear Power Plant. The proposed amendment would revise the Technical Specification 5.5.6 Primary Containment Leak Rate Testing Program to allow a permanent extension of the Type A Primary Containment Integrated Leak Rate Test interval to 15 years and allow an extension of Type C Local Leak Rate testing interval up to 75 months.

The Nuclear Regulatory Commission staff is reviewing the submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). On October 21, 2016, the Entergy staff indicated that a response to the RAI would be provided within 30 days of the date of this letter.

Please contact me at (301) 415-3629 if you have any questions on this issue.

Sincerely,

Diane Render, Ph.D.
Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure:
As stated

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING LICENSE AMENDMENT REQUEST TO REVISE

TECHNICAL SPECIFICATIONS SECTION 5.5.6

ENTERGY NUCLEAR FITZPATRICK, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

In an application dated August 29, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16242A332), Entergy Nuclear Operations, Inc. (Entergy), submitted an application for a proposed amendment to the Technical Specifications (TSs) for James A. FitzPatrick Nuclear Power Plant (FitzPatrick). The proposed amendment would revise the TS 5.5.6 Primary Containment Leak Rate Testing Program to allow a permanent extension of the Type A Primary Containment Integrated Leak Rate Test interval to 15 years and allow an extension of Type C Local Leak Rate testing interval up to 75 months. The Nuclear Regulatory Commission staff is reviewing the submittal and has the following questions:

SBPB

RAI-1

The license amendment request (LAR) page 32 states "In accordance with TS 5.5.6, the allowable maximum pathway total Types B and C leakage is 0.6 La where La equals 320 SLM". Thus the performance criterion of 0.6 La for evaluating combined type B and C leakage would be 192 SLM. However, TS 5.5.6 states "During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 La for the Type B and Type C tests." NEI 94-01 Section 8.0 states:

> [American National Standards Institute/American Nuclear Society] ANSI/ANS 56.8-1994 also specifies **surveillance acceptance criteria for Type B and Type C tests**. The ANSI/ANS 56.8-1994 **definition is that the combined leakage rate for all penetrations subject to Type B or Type C tests is limited to less than or equal to 0.6 La, when determined on a [Minimum Pathway Leak Rate] MNPLR basis from As-found LLRT results**; and limited to less than or equal to 0.6 La, as determined on a Maximum Pathway Leakage Rate (MXPLR) basis from the As-left LLRT results.

Enclosure

> Due to the performance-based nature of Option B to Appendix J and this guideline, it is recommended that **acceptance criteria for the combined As-found leakage rate for all penetrations subject to Type B or Type C testing be the same as that defined in ANSI/ANS 56.8-1994** with the following additions. The combined As-left leakage rates determined on a MXPLR basis for all penetrations shall be verified to be less than 0.6 La prior to entering a mode where containment integrity is required following an outage or shutdown that included Type B and Type C testing only. **The combined As-found leakage rates determined on a MNPLR basis for all penetrations shall be less than 0.6 a at all times when containment integrity is required**.

TS 5.5.6 indicates that the leakage rate testing program will be in accordance with Regulatory Guide (RG) 1.163 which endorsed Nuclear Energy Institute's (NEI's) 94-01. Determining the as-found minimum pathway combined Type B and C test totals verifies that the requirement had been met at all times when containment integrity was required. As left values are a permissive and can always be achieved through more maintenance while as-found values demonstrate acceptable performance is being maintained throughout the plant operating cycles.

Table 3.4.5-1 shows a row for the outage totals as "Fraction of La" and that for 2012 RO20 as found minimum pathway that fraction was 0.9315. There was no explicit indication in the LAR that this represented a failure to meet the surveillance performance criterion or that past operability of the primary containment had been evaluated given how close the Type B and C total was to the overall primary containment performance criterion of La and that the combined Type B and C leakage did not necessarily account for all containment leakage potential.

a. Was there a failure to meet the surveillance acceptance criterion (0.6 La) for as-found Type B and C total in 2012?
b. Was the possibility for overall containment leakage potential to have exceed the La criterion evaluated? If not, at what value of as-found minimum pathway combined Type B and C leakage would past containment operability be evaluated?
c. Justify why the requested test interval extensions should be considered prudent given the 2012 RO-20 and 2014 RO-21 combined Type B and C minimum pathway test results of 0.9315 La and 0.503 La respectively?

RAI-2

The LAR page 66 indicates that an adjustment will be applied for those Type C tested components on a 75-month interval. The NEI 94-01 Revision 3 Condition 2 indicated that an estimate of the test total understatement be made for LLRT valve testing intervals beyond 60-months.

a. When the LAR refers to Type C tested components on a 75-month interval, does this include all Type C tested components scheduled for intervals exceeding 60-months, which would include those that may be scheduled for intervals between 60-months and 75-months?

APLA

RAI-1

Section 4.2.7 of Electric Power Research Institute (EPRI) Topical Report (TR)-1009325, Rev. 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325" states that "[w]here possible, the analysis should include a quantitative assessment of the contribution of external events (for example, fire and seismic) in the risk impact assessment for extended [Integrated Leak Rate Test] ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event."

The "Fire Analysis" section of Attachment 1 to the LAR states that "although the JAF Individual Plant Examination of External Events (IPEEE) fire risk model has not been updated since its original issuance, use of the IPEEE model would tend to give conservative results."

A fire PRA model with sufficient quality and detail that has undergone a peer review under American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-SA-2009, "Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", as clarified/qualified by Rev. 2 of Regulatory Guide 1.200 (RG 1.200), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," is not utilized for this application. The IPEEE model was issued 20 years ago in June 1996, and there have been significant changes to Fire PRA methodology since then.

a. Identify any gaps between the FitzPatrick IPEEE fire risk model and the "Internal Fire Technical Elements" required by Rev. 2 of RG 1.200 that are relevant to this submittal and explain why these gaps do not have significant impact on the risk assessment results for this application.

   OR

   Demonstrate that addressing the requirements would have no impact on the final results in the LAR by providing a sensitivity evaluation to show that the use of the IPEEE results remains conservative—replace the IPEEE results, including at least the following:

   i. Reanalysis of all containment bypass scenarios leading to LERF from the internal events PRA, replacing any random failure probabilities with appropriate fire-induced values (e.g., NUREG/CR-7150 values for MOVs for ISLOCAs) - this should include any ISLOCA pathways previously screened out due to low random failure probability, as the probability could be significantly higher if fire induced.
   ii. Reanalysis of the IPEEE fire LERF using updated fire frequencies and, if suppression was credited, updated non-suppression probabilities from NUREG-2169.

The "Seismic Analysis" section of Attachment 1 to the LAR states that "In the IPEEE, JAF [FitzPatrick] performed a Seismic Margin Assessment (SMA), which is a deterministic and conservative evaluation that does not calculate risk on a probabilistic basis." This section does not cite the use of the results from Generic Issue (GI)-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment." It implies that there was no seismic quantification or sensitivity analysis performed for the application. However, Section 4.0 of Attachment 4 to the LAR indicates that a sensitivity analysis was performed based on GI-199.

b. Provide additional details in the "Seismic Analysis" section of the LAR to clarify that an estimate from GI-199 based on the United States Geological Survey (USGS) seismic hazard curves was used for quantification (bounding CDF = 6.1E-6/y from "Weakest Link Model" stated in GI-199) performed in Attachment 4 of the LAR.

RAI-2

Section 1.0 of Attachment 4 to the LAR states that "the purpose of this analysis is to provide a risk assessment of permanently extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the James A. FitzPatrick Nuclear Power Plant (JAF). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A [Reference 1]."

Section 2.0 of Attachment 4 to the LAR states that "the basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessment of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. This section does not contain the basis for extending the Type A Primary Containment ILRT interval to 15 years.

Provide the basis for extending the ILRT to a 15-year test interval per NEI 94-01, Rev. 3-A.

RAI-3

Section 5.1.2, Table 5-1 and Table 5-2 of Attachment 4 to the LAR contain the summary of the Internal Events core damage frequency (CDF) and large early release frequency (LERF) results, respectively, for the FitzPatrick PRA Model when quantified using the Level 2 model top gates. For the loss-of-coolant accident "LOCA Outside Containment" internal event, the LERF (1.09E-08) is lower than the CDF (1.78E-08). If there is no release mitigation credited for that internal event, the CDF and LERF values would be expected to be the same.

What was credited in the Probabilistic Risk Assessment (PRA) model that contributed to a lower LERF value for the "LOCA Outside Containment" internal event? Verify the credits taken reflect the "as-designed, as-built, as-operated" plant configurations per RG 1200, Rev. 2.

RAI-4

Section 4.2.2 of EPRI TR-1009325, Rev. 2-A states that "The most relevant plant-specific information should be used to develop population dose information. The order of preference shall be plant-specific best estimate, Severe Accident Mitigation Alternative (SAMA) for license renewal, and scaling of a reference plant population dose."

Section 5.2.2 of Attachment 4 to the LAR states that the population doses were calculated using the methodology of scaling Peach Bottom population doses to JAF and the scaling produced more conservative population dose values. JAF's plant-specific data is not used in calculating the population doses, while more recent plant-specific data exists.

Provide justification of (1) why plant-specific data is not used and (2) why using Peach Bottom's data would produce a more conservative value. Is the data of the two sites sufficiently comparable to render this scaling approach based on power level, leakage, and population conservative?

RAI-5

In Section 5.2.2 of Attachment 4 to the LAR, Table 5-9 lists the population dose for Class 7a, "Severe accident phenomena induced failure (early)" is "1.28E+06 person-rem," while the population dose for Class 7b, "Severe accident phenomena induced failure (late)" is "6.81E+05 person rem." When used in conjunction with the release class frequencies for Class 7a and Class 7b in Table 5-10, the annual population dose resulting from late containment failure (1.01 person-rem per year) is higher than that resulting from early containment failure (0.687 person-rem per year).

In NUREG-1437, "Generic Environment Impact Statement for License Renewal of Nuclear Plants", Table 5-4 and Table G-2 show that the annual population doses due to early containment failure and late containment failure are 0.87 person-rem per year and 0.76 person rem per year, respectively. That is, the value for early containment failure exceeds that for late containment failure—a trend opposite to the one in the LAR.

Explain the differences between the NUREG-1437 and the LAR results; provide references for supporting assumptions and calculations.

RAI-6

In Section 5.3.1.of Attachment 4 to the LAR, the fire LERF result was calculated based on the fire CDF (2.12E-5) taken from the original FitzPatrick IPEEE, "James A. FitzPatrick Nuclear Power Plant Individual Plant Examination of External Events", which was published in June, 1996. However, an updated fire CDF (3.42E-5) that is higher than the one listed in the original IPEEE is available in Section 2.1.7 of "James A. FitzPatrick Nuclear Power Plant - Review of FitzPatrick Individual Plant Examination of External Events (IPEEE) Submittal (TAC NO. M83622)" dated September 21, 2000 ADAMS Accession No. ML003743771). Using a smaller fire CDF value from the original IPEEE to obtain the fire LERF produced a less conservative fire LERF value in the LAR.

Subsequent to resolving the question in RAI 1.a above regarding the conservatism when using the

FitzPatrick Fire IPEEE risk, confirm that the total change in LERF is still within the "very small" and "small" risk increase regions of Regulatory Guide 1.174. That is, perform the calculations in Section 5.3 using the updated fire CDF value and correct all subsequent calculations in the LAR that are affected by using the non-conservative old fire CDF value (2.12E-5).

RAI-7

In Table 5-21, Section 5.3.2 of Attachment 4 to the LAR, the dose rate for dose rate increase from 3-per-10 years to 1-per-10 years is 3.76E-03; the dose rate for dose rate increase from 3-per-10 years to 1-per-15 years is 6.45E-03; the dose rate for dose rate increase from 1-per-10 years to 1-per-15 years is 2.69E-03. However, based on Table 5-13, these three increases should be 5.08E-3, 8.71E-3 and 3.63E-3, respectively, which are higher than those listed in Table 5-21.

Correct the dose rates in Table 5-21 and all affected subsequent calculations in the LAR based on the correct dose rates listed in Table 5-13 to ensure the affected calculations do not change the final results of this LAR.

RAI-8

Section 5.3.3 of Attachment 4 to the LAR states that "Taking the baseline analysis and using the values provided in Tables 5-10 and 5-10 for the expert elicitation sensitivity yields the results in Table 5-23."

Confirm that the 2nd "5-10" in the above sentence is a typo and it should be replaced by "5-22".

RAI-9

In Table 6-1, Section 6.0 of Attachment 4 to the LAR lists the dose values for Class 1, 3a and 3b are 9.15E+02, 9.15E+03 and 9.15E+04, respectively. In Table 5-10, Section 5.2.3 of Attachment 4 to the LAR, the dose values for Class 1, 3a and 3b are 2.97E+03, 2.97E+04 and 2.97E+05, respectively.

The values in Table 6-1 and Table 5-10 are inconsistent. Justify the lower dose values used in Table 6-1, Section 6.0 of Attachment 4 to the LAR; or if the dose values in Table 5-10 are correct, replace Table 6-10 dose values with the values from Table 5-10 and recalculate all subsequent results as necessary.

RAI-10

Address the following in the F&Os section in Table A-1 of Attachment 1, "PRA Technical Adequacy" to the LAR:

a. Finding (Page 43 of 53 in the LAR, Attachment 4): "Appendix H4 states that when applying the dependency model, the dependency was generally applied to the Human Error Probability (HEP) with the higher HEP (versus assigning the dependency to the action which occurs later in time). This does not conform to the established calculation method. In general, it is more appropriate to choose the event

that would occur first in the accident sequence..."

To ensure the final CDF and LERF results remain conservative and are not affected by some inadequately low joint HEP values, JAF is expected to follow the requirement of HR-G7 listed in Table 2-2.5-8(g) in ASME/ANS RA-Sa-2009 to assess the HEP dependencies in the cutsets that contain multiple HEPs. Although NRC has not endorsed any standards regarding the floor value of a joint HEP, Section 2.4.3.5 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)" states that "The total combined probability of all the HFEs in the same accident sequence/cut set should not be less than a justified value. It is suggested that the value not be below ~0.00001 since it is typically hard to defend that other dependent failure modes that are not usually treated (e.g., random events such as even a heart attack) cannot occur. Depending on the independent HFE values, the combined probability may need to be higher." EPRI TR-1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a basis for extending this lowest limit by an order of magnitude to 1E-6, which may be characteristic only for internal events.

The status of this finding is still open because "the new HRA guidance has not yet been incorporated into the JAF PRA model." The LAR states that this finding is "not significant", since "the assigned dependencies are often conservative because they are based on the overall HEP value (whereas dependency between the execution portions of the HEPs can frequently be justified as being low or zero). Re-analysis of the combined HEPs to address this Fact and Observation (F&O) resulted in a CDF increase of 1.73% and a LERF increase of 0.02%..."

Regarding the re-analysis of the combined HEPs, was there any joint HEP floor value applied to the cutsets that contain multiple HEPs? If so, and if the floor value is significantly less than the value stated in NUREG-1792 or EPRI TR-1021081, provide justification.

If there is no floor value applied to those cutsets that contain multiple HEPs, provide justification and a sensitivity evaluation using a joint HEP floor value of ≥1E-6 to show that these low joint HEP values have no significant impact on the overall results stated in the LAR.

b. Finding (Page 44 of 53 in the LAR, Attachment 4): "The JAF PRA determined a point estimate for damage and documented an analysis for parametric uncertainty. However, the state of knowledge correlation was not fully accounted for due to the manner in which the JAF basic event data base is constructed."

The SR description for this finding did not include LERF. However, the QU-A3 requirement listed in Table A-2 of RG 1.200 states that "The state-of-knowledge correlation should be accounted for all event probabilities. Left to the analyst to determine the extent of the events to be correlated. Need to also acknowledge LERF quantification."

Provide the quantitative results from the sensitivity evaluation performed at the time

of the RG 1.200, Rev. 2 peer review—the results should include the effect on both CDF and LERF due to parametric/state of knowledge uncertainty.

c. Finding (Page 45 of 53 in the LAR, Attachment 4): "The level 2 model uses point estimates for success branches while the failure logic is modeled for the failure branch."

   The LAR states that "no impact on the application is expected" because the "use of point estimates is reasonable for top events that include phenomena, and system related success probabilities are approximately equal to 1.0.

   Provide the values of all success probabilities to show that they are sufficiently close to 1.0 and have minimal impact on the quantification.

d. Finding (Page 46 of 53 in the LAR, Attachment 4): "Spray-induced and submergence induced failures appear to have been addressed in the analysis. No documentation of a systematic assessment of the effects of jet impingement, pipe whip, humidity, temperature, etc., on structure, system or components (SSCs) could be identified. No evaluation of the specific equipment evaluated in the PRA compared to equipment considered in the design analyses, e.g., EQ lists was documented. Since PRA can credit non-safety-related equipment, relying on design basis evaluations to dismiss these dynamic effects may credit equipment that cannot withstand the effects considered in the design analysis. In addition, failure in a system containing high temperature fluid can actuate fire systems and impact additional equipment. Also, the PRA models may evaluate breaks beyond those of the design basis."

   The status of this finding is open and the licensee states that "The only requirement to meet Capability Category II (CC-II) is to document that those mechanisms are not included in the scope."

   However, Table A-3 in Rev. 2 of RG 1.200, IFSN-A6 requires that "For Cat II, it is not acceptable to just note that a flood-induced failure mechanism is not included in the scope of the internal flooding analysis. Some level of assessment is required." Also, "For the SSCs identified in IFSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions."

   Provide the results from the analysis related to IFSN-A6 to show that CC-II requirements are met; or explain how meeting only CC-I requirements does not have impact on the final conclusions of this application.

Please contact me at (301) 415-3629 if you have any questions on this issue.

Sincerely,

/RA/

Diane Render, Ph.D.
Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

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