NRR-PMDAPEm Resource

From: Kim, James
Sent: Monday, May 19, 2014 10:58 AM
To: 'Couture III, Philip' (pcoutur@entergy.com)
Cc: cchappe@entergy.com
Subject: Vermont Yankee RAI for LAR on Eliminate Certain ESF Requirements During Movement of Irradiated Fuel (TAC No. MF3068)

Phil,

By letter dated November 14, 2013, Entergy Nuclear Operations, Inc. (the licensee) submitted an amendment request to eliminate operability requirements for secondary containment when handling sufficiently decayed irradiated fuel and while performing core alterations.

The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has determined that additional information is needed to complete its review. The specific questions are shown below in the request for additional information (RAI). Based on our discussions on May 8, 2014, we understand that responses to the RAI questions will be provided by June 9, 2014. These RAI questions will be made publicly available.

Thanks,

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

TECHNICAL SPECIFICATION TASK FORCE TRAVELER 51

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

By application dated November 14, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13323A518), Entergy Nuclear Operations submitted a license amendment for Vermont Yankee (VY). The proposed license amendment request (LAR) would eliminate operability requirements for secondary containment when handling sufficiently decayed irradiated fuel and while performing core alterations using Technical Specification Task Force (TSTF) – 51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations."

RAI 1

Attachment 4, Table 3-2, entitled "VYNPP [VY Nuclear Power Plant] – Re-analysis of AST/FHA [alternative source term/fuel handling accident] Radiological Consequences with Open Containment" (ADAMS Accession No. ML13323A519) of the November 14, 2013 application, provides a core inventory based upon a core average maximum burnup of 58 giga-watt-days per metric ton of uranium (GWD/MTU). Attachment 4, Table 3-

1 states that the FHA uses Table 3 gap fractions from Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Adams Accession Number ML003716792). Footnote 11 for Table 3 of RG 1.183 states that Table 3 is acceptable for use with currently approved reactor light water fuel with a peak burnup of up to 62,000 mega-watt-days per metric ton of uranium (MWD/MTU) (equivalent to 62 GWD/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatts per foot (kW/ft) peak rod average power for burnups exceeding 54,000 MWD/MTU (equivalent to 54 GWD/MTU). Since the assumed fuel burnup for the Attachment 4, Table 3-2 core inventories appear to exceed the RG 1.183, footnote 11 limits, please confirm that the VY fuel burnup and linear heat generation rates comply with footnote 11. If not, please justify the use of Table 3 from RG 1.183 with fuel outside the burnup and linear heat generation rates used to derive Table 3.

RAI 2

Page 9 of 17 of the application, entitled "Technical Specifications Proposed Change No. 306, Eliminate Certain ESF [Engineered Safety Feature] Requirements during Movement of Irradiated Fuel," dated November 14, 2013 (ADAMS Accession Number ML13323A518) states:

> *The accidents postulated to occur during core alterations, in addition to the fuel handling accident, are* [the] *inadvertent criticality due to control rod removal error and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Therefore, the only accident postulated to occur during core alterations that result in significant radioactive release is the FHA* [fuel handling accident]. *Thus, the consequence of a FHA envelops the consequences of potential accidents postulated to occur during core alterations.*

Page 14 of 17 of the application also states that the proposed changes follow Technical Specification Task Force traveler 51 (TSTF-51), Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations" (ADAMS Accession Number ML040400343).

TSTF-51 states:

> *The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by analysis* [emphasis added] *that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10 CFR 100)* [or 10 CFR 50.67].

Standard Review Plan (SRP) 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," (ADAMS Accession Number ML003734190) states:

> *The models, assumptions, and parameter inputs used by the licensee should be reviewed to ensure that the conservative design basis assumptions outlined in RG-1.183 have been incorporated.*

Appendix B of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession Number ML003716792), Regulatory Position 1.1 states:

> *The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case.*

After reviewing the information submitted by the VY submittal to adopt TSTF-51, the NRC staff needs additional information to verify that the limiting cases have been considered.

a. Please provide a FHA analysis that evaluates the dropping of loads allowed over irradiated fuel assemblies (i.e. sources or reactivity control components) onto irradiated fuel assemblies with 24-hours

of decay time. The analysis should only credit those safety systems required to be operable as required by technical specification [TS]. This will provide the staff with reasonable assurance that the FHA doses remain within regulatory limits when references to Core Alterations are removed from TSs and ESFs are no longer required during movement of loads such as sources or reactivity control components.

b. Page 7 of 17 of the application dated November 14, 2013 states that two main configurations of the Reactor Building during fuel movement were considered. The second configuration discusses "various [emphasis added] pre- and post-FHA Main Control Room (CR) ventilation configurations that would support refueling with open containment," but does not define which configurations are credited in the proposed TS changes. These ventilation configurations are discussed in the submittal, but the NRC staff needs some clarification regarding these configurations. Please state the proposed new design basis configuration credited to support the TS changes.

RAI 3

Page 10 of 17 of the application, dated November 14, 2013, states:

> *The operability requirements during movement of a fuel cask for ESF mitigation are deleted as part of this proposed license amendment.*

and,

> *Since the FHA resulting from a dropped fuel cask is shown to not be credible, the proposed TS changes omitting operability requirements during movement of a fuel cask ESF mitigation is justified.*

SRP 15.7.5, "Spent Fuel Cask Drop Accidents," (ADAM Accession No. ML052350315) states:

> *A design basis radiological analysis is performed if a cask drop exceeding 30 feet can be postulated or if limiting devices are removed during cask handling within the plant so the 30-foot drop height is exceeded. If the radiological consequences of a cask drop accident are to be computed, then information on whether building leaktightness can be expected after a cask drop is obtained from ASB* [Auxiliary Systems Branch] *(e.g., whether the technical specifications require large doors to be closed during fuel handling or whether ventilation systems should be operating and whether the building leaktightness would be violated by the cask drop).*

At VY can a spent fuel cask drop exceed 30 feet or can the limiting devices be removed during cask handling? If so, please provide the radiological consequences of a cask drop accident. Please justify all answers.

RAI 4

Page 9 of Attachment 4 of the application, dated November 14, 2013, states that the activity releases from the containment atmosphere over two hours is 98.2%. Appendix B of RG 1.183, Regulatory Position 5.3 states if the containment is open during fuel handling, the radioactivity that escapes from the reactor cavity pool to the containment is released to the environment over a two-hour time period. RG 1.83, Regulatory Position 5.1.2, "Assignment of Numeric Input Values," states that the numeric values that are chosen as inputs to the required analysis should be selected with the objective of determining a conservative dose. Please justify why a conservative value of 100% of the activity in containment was not assumed to be released from the containment over the two hour time period.

RAI 5

Page 10 of Attachment 4 of the application, dated November 14, 2013, states that four sensitivity cases make use of several rates to assess the dose impact on the main CR purge initiation time. Please describe which case is to be reviewed for the design basis and clarify what is meant by the "purge initiation time."

RAI 6

Regulatory Position 5.1.2 of RG 1.183 states: "The single active component failure that results in the most limiting radiological consequences should be assumed." State the most limiting single active failure for FHA and justify the answer.

RAI 7

Attachment 4, Table 3-1, dated November 14, 2013, states that VY assumes an overall pool decontamination factor (or DF) of 200 based upon Appendix B of RG 1.183. The DF of 200 is based upon reference B-1 ("Evaluation of Fission Product Release and Transport," (ADAMS Accession No. 8402080322)) of RG 1.183. The data upon which the pool DF of 200 is based was developed in 1971 and was based on the Westinghouse fuel marketed at the time (the assumed internal fuel pressure of 1200 pounds force per square inch gage (psig) was used). Since higher pressures correlate to lower DFs, the NRC staff would like VY to confirm that the fuel VY uses will have an internal fuel pressure of less than 1200 psig. If not, please provide the experimental data for current fuel types used at VY that justify a DF of 200 for fuel pressures greater than 1200 psig. Also, please provide a detailed justification for using a DF of 200 for pressures up to 1200 psig.

RAI 8

Please provide a justification for all changes from the current licensing basis (See Issue 1 of NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," (ADAMS Accession No. ML053460347) for more detail). No justification is needed for changes that are consistent with Regulatory Guide 1.183 or are provided in the submittal dated August 13, 2013 (ADAMS Accession No. ML13247A076) unless requested by these RAIs.

RAI 9

The changes to TS 3.9.4 allow an "open" containment when moving fuel that is not recently irradiated. Consistent with Regulatory Issue Summary 2006-04, please confirm that all pathways to the environment created by the proposed changes are considered and analyzed in the FHA analysis.

a. Please confirm that the most limiting combination of release point and receptor for the control room were used to determine atmospheric dispersion factors for each accident.

b. State and justify the release points that correlate to the atmospheric dispersion factors used.

RAI 10

SRP 16.0, "Technical Specifications," (ADAMS Accession No. ML100351425) states: "In TS change requests for facilities with TS based on previous STS [Standard Technical Specifications], licensees should comply with comparable provisions in these STS NUREGs to the extent possible or justify deviations from the STS." Please provide a justification for deviations from the STS created by the proposed changes.

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