OFFICE OF NUCLEAR REACTOR REGULATION

REGULATORY AUDIT TOPICS REGARDING THE

REACTOR DESCRIPTION, RADIATION PROTECTION, AND ACCIDENT ANALYSIS

DESCRIBED IN THE APPLICATION FOR LICENSE RENEWAL

LICENSE NO. R-33

GENERAL ELECTRIC-HITACHI NUCLEAR ENERGY AMERICAS LLC

DOCKET NO. 50-073

By letter dated November 19, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21053A071), GE-Hitachi Nuclear Energy Americas LLC (GEH) applied for renewal of Facility Operating License No. R-33 for the Nuclear Test Reactor (NTR) in accordance with the requirements contained in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The requested licensing action would renew the facility operating license for a period of 20 years.

During the review of GEH's license renewal application (LRA), the U.S. Nuclear Regulatory Commission (NRC) staff identified additional information needed to continue its review of the reactor description, radiation protection, and accident analysis, as described in the NTR final safety analysis report (FSAR).

To support the review of the NTR LRA, the NRC staff will conduct a virtual regulatory audit on July 28, 2021, and August 4, 2021, to confirm the information in the NTR FSAR. The specific topics below identify areas where additional information is needed for the NRC staff to continue its review of the NTR LRA and prepare a safety evaluation report. Specific chapters and technical areas of the NTR FSAR to be reviewed include the following:

- Chapter 3, "Design of Structures, Systems, and Components"
- Chapter 4, "Reactor Description"
- Chapter 10, "Experimental Facilities and Utilization"
- Chapter 11, "Radiation Protection Program / Waste Management"
- Chapter 13, "Accident Analyses"

Regulatory Basis and Applicable Guidance Documents

The NRC staff is evaluating the NTR LRA using the following regulations and guidance:

- Section 50.34, "Contents of applications; technical information," paragraph (b)(2) of 10 CFR requires, in part, that an FSAR include a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

- Section 50.9, "Completeness and accuracy of information," of 10 CFR require that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

- Part 20, "Standards for Protection against Radiation," of 10 CFR require that doses to workers and members of the public be limited.

The NRC staff's review of the NTR LAR is also based on the following:

- NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," issued February 1996 (ADAMS Accession No. ML042430055)

- NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," issued February 1996 (ADAMS Accession No. ML042430048)

In particular, the requested information in this regulatory audit is intended to support the NRC staff in making appropriate evaluation findings based on the following:

- NUREG-1537, Part 2, "General Requirements," states, in part:

  The SAR contains the formal documentation for a facility, presenting basic information about the design bases, and the considerations and reasoning used to support the applicant's conclusion that the facility can be operated safely. The descriptions and discussions therein also support the assumptions and methods of analysis of postulated accidents, including the maximum hypothetical accident (MHA), and the design of any engineered safety features (ESFs) used to mitigate accident consequences.· The MHA, which assumes an incredible failure that can lead to fuel cladding or to a fueled experiment containment breach, is used to bound credible accidents in the accident analysis.

- Section 10.2, "Experimental Facilities," of NUREG-1537, Part 2, states, in part:

  The applicant should show that the design of experimental facilities does not introduce new mechanisms to initiate a LOCA, or that the potential consequences of a LOCA caused by the failure of experimental facilities are considered acceptable by the Chapter 13 analysis. The applicant should show that fuel integrity would not be lost as a result of a LOCA initiated or affected by the design of an experimental facility or by the malfunction of an experiment or experimental facility.

- Chapter 13, "Accident Analyses," of NUREG-1537, Part 1, states, in part:

  The applicant should submit information and analyses that show that the health and safety of the public and workers are protected and that the applicant has considered potential radiological consequences in the event of malfunctions and the capability of the facility to accommodate such disturbances [and] that the facility, design features, safety limits, limiting safety system settings; and limiting conditions for operation have been selected to ensure that no credible accident could lead to unacceptable radiological consequences to people or the environment.

- Chapter 13, “Accident Analyses,” of NUREG 1537, Part 2, states, in part:

  The information in this chapter should [demonstrate] that all potential accidents at the reactor facility have been considered and their consequences adequately evaluated. Each postulated accident should be assigned to one of the following categories, or grouped consistently according to the type and characteristics of the particular reactor.

  - MHA
  - insertion of excess reactivity (ramp. step, startup, etc.)
  - loss of coolant
  - loss of coolant flow
  - mishandling or malfunction of fuel
  - experiment malfunction ·
  - loss of normal electrical power
  - external events
  - mishandling or malfunction of equipment

- Section 3.1, “Design Criteria” of NUREG 1537, Part 2, states, in part:
  Areas of review should include the criteria for the design and construction of the structures, systems, and components that are required to ensure … safe reactor operation, safe reactor shutdown and continued safe conditions, [and] response to potential accidents analyzed in Chapter 13, “Accident Analyses,” of the SAR.

<u>Audit Topics</u>

The NRC staff requests the following information to support its LRA review:

1. In Section 4.2, the reactor fuel container is described as being in service since 1976. Describe how the container is maintained or tested to continue to ensure no leaks.

2. FSAR Section 4.2.2 describes a latching mechanism by which poison sheets are attached to the core. Describe this mechanism in further detail and explain why poison sheets would not move relative to the core during a seismic event.

3. FSAR Section 4.4.2 states that the coolant temperature coefficient is calculated based on overall temperature changes and describes determination of the void coefficient by extrapolating the temperature coefficient data. FSAR Figure 13-19 shows that some void fraction is expected for steady-state operation at all non-zero power levels.

   a. Provide the experimental data and calculations of the coolant temperature and void coefficients of reactivity.

   b. Based on the discussion in this same paragraph, the NRC staff interprets that the coolant temperature coefficient includes effects of both water density change and void buildup, and that the void coefficient of -5.7 ¢/% void is only applicable over 124 Fahrenheit.

i. Is the NRC staff interpretation correct? If not, explain the effects included in the coolant temperature coefficient.

ii. Explain over what ranges the reactivity coefficients listed in Table 4-2 are applicable.

4. In Section 4.4.3, the FSAR states that manual poison sheets shall be restrained in a manner which will prevent movement by more than ½ inch relative to the reactor core.

   a. How much movement is expected once the poison sheets have been latched in place?

   b. Is this potential movement in the radial or azimuthal direction?

   c. Discuss the effect of this movement on the reactivity worth of the poison sheets.

5. In Section 4.4.3, the NTR FSAR provides rod worths.

   a. Describe how the total safety rod worth of $3.86 is determined.

   b. Additionally, explain which rod worths listed in Chapter 4 were determined from experiment, and which were determined from model predictions.

6. FSAR Section 4.4.4, states, in part, that "differences between the core model-predicted control rod worths and net reactivity gains from MPS changes and the NTR measured data are within the overall model uncertainty…."

   a. Does this apply to safety rod worths?

   b. What is considered in calculation of the overall model uncertainty?

7. FSAR Section 5.3 discusses primary to secondary coolant water leakage. Describe the actions taken if a heat exchanger tube leak is detected and how that affects the heat transfer capabilities.

8. In FSAR Section 11.1.1, GEH describes the use of models to show the ratio of noble gases to Iodine-131 to determine the amount of Argon-41 produced at the NTR facility.

   a. The NRC staff would like to review and discuss the calculations that supports this information in the reading room to understand the method(s) used to verify this relationship, including the acceptability of the ratio used by GEH to determine the Argon-41 produced at the facility.

   b. The NRC staff also request to view the Radioisotope Buildup and Decay [RIBD] calculation and discuss how GEH plotted the data to show the relationships.

9. FSAR Section 11.1.2.3 briefly talks about the radiation protection training at the GEH facility. The NRC staff requests a list of the radiation safety training be made available for review in the reading room to provide a general understanding of the training available at the GEH NTR facility.

10. For the NTR accident analyses, consider providing a table with limiting values (safety limit vs. calculated values) for easier reference.

11. FSAR Chapter 13 describes some postulated accidents, however, other categories of accidents described in NUREG-1537, Part 2 are not evaluated and analyzed.

    a. Identify the postulated MHA analyzed and evaluated dose consequence for workers (licensee staff) and members of the public.

    b. Revise FSAR Chapter 13 to describe how the postulated accident categories in NUREG-1537, Part 2 are analyzed and evaluated for the GEH NTR.

    c. For each applicable postulated accident category, identify the limiting event selected for detailed quantitative analysis, evolution of the scenario, and address the likelihood of occurrence.

12. FSAR Section 13.1, states, in part, "Release of radioactive materials to the environs that result in exceeding the limits of 10 CFR 20.1301/20.1302 for members of the public," is an unacceptable consequence of anticipated operational occurrences, but not as an unacceptable consequence for postulated accidents. Which (if any) postulated accidents have this consequence?

13. In FSAR Section 13.3.1, verify that, if there is a loss of power, the magnet power supply would deenergize, causing loss of power to the safety rod electromagnets and the safety rods would insert into the core by their spring action.

14. FSAR Section 13.3.5 states, in part, that "[t]he physical arrangement of the fuel container is such that an element located in the loading chute results in a worse core geometry than the cylinder formed by having all elements in the core support reel." In this context, explain if "worse core geometry" means a geometry with a smaller multiplication factor.

15. FSAR Figure 13-3 shows the fuel melting temperature to be "1050 °F," but Figure 13-4 shows the melting temperature to be greater than "1100 °F." Provide the actual melting temperature of the fuel.

16. FSAR Section 13.4.3 discusses peak fuel temperatures, but only peak cladding temperatures are depicted. Provide the peak fuel temperatures for these transients.

17. FSAR Section 13.4.3 considers various reactivity insertions. Explain if these reactivity insertions include reactivity added by coolant temperature changes over the course of the transient, e.g., assuming the reactivity inserted by coolant temperature increase is $0.15, is the initial reactivity step $0.61 (such that the total reactivity insertion is $0.76) or is the initial reactivity step $0.76 (such that the total reactivity insertion is $0.91)?

18. The title of FSAR Figure 13-14 is “LOCA – **PCT** (°F)”, [emphasis added] but the caption is “**Fuel Temperature** Following Loss of Coolant Accident.” The preceding discussion states that the figure depicts fuel temperature. Explain the parameter that is plotted in Figure 13-4.

19. FSAR Section 13.4.6 describes an assumed convection heat transfer coefficient on the exposed surface of the reactor. Does this refer to the outer surface of the graphite cube?

20. FSAR Section 13.6.3 and 13.6.4, and the “Environmental Information Report for the General Electric Test Reactor,” discuss computer codes used at the GEH facility. The NRC staff requests the following be made available for review in the reading room:

    a. Input/output files from ORIGEN2 computer code version 2.1 used to calculate the fission products created during operation of the uranium-235 capsule (FSAR Section 13.6.3 Calculation Method).

    b. Input/output files from RADTRAD computer code version 3.10 used to calculate the total effective dose equivalent dose resulting from exposure from an accidental release of the uranium-235 capsule to workers (licensee staff) and members of the public (FSAR Section 13.6.3 “Calculation Method” and FSAR Section 13.6.4 “Results”).

21. FSAR Section 13.7.4 states that a scram delay time of 0.200 seconds is assumed in transient analyses.

    a. How was the scram delay time of 0.200 seconds determined?

    b. Explain the portion of this delay that is from instrumentation and the portion that is from physical movement of the control rods.

    c. How does this delay compare to measured delay time?

22. FSAR Section 13.7.3.1, states, in part, that “DNB heat flux correlation is applicable to low-velocity, low-pressure, saturated boiling conditions with a significant void fraction.” Is this state expected to occur in any of the reactor operating conditions or testing conditions?

23. FSAR Section 16.1 states that the fuel cladding thickness is half of its original value. Discuss and describe the limiting thickness where a fuel disk would need to be replaced.

24. In DBR-0056267, “Submittal of NTR MCNP Inputs and Descriptions,” Section 1.2, “MCNP Input Changes for Rod Worth Calculations,” it describes that the total worth of control rods is determined with safety rods fully withdrawn and no manual poison sheet (MPS) inserted, and that the total worth of safety rods is determined with no MPS inserted and control rods withdrawn.

    a. How would insertion of MPS affect total control rod worth?

    b. How would insertion of MPS affect total safety rod worth?

    c. How would positioning of control rods at critical positions affect total safety rod worth?

25. In reference to “Submittal of NTR TRACG Basedeck”:

    a. Explain the nodalization used for the vessel Section 1.2.

        i. How were 8 axial segments determined to be appropriate?

        ii. Are all the axial segments identical in size?

        iii. Are there any reasons a finer nodalization should be used for the core region?

    b. Discuss if experimental data is used Section 1.3 to tune the model of the heat exchanger.

    c. In Section 1.4, describe if any airgap expected in GEH NTR fuel.

26. Proposed TS 1.2.15 defines “Potential Excess Reactivity” as “That excess reactivity which can be added by the remote manipulation of control rods plus the maximum credible reactivity addition from primary coolant temperature change plus the reactivity worth of all installed experiments.”

    a. Provide an example calculation to show how potential excess reactivity is calculated in order to meet TS 4.1.3.1.

    b. Describe how changes in graphite temperature factor into the excess reactivity calculation and the change in graphite temperature that is expected during normal operation.

    c. Could reactivity be added by other means of control rod movement exceed that added by “remote manipulation”?

27. Proposed TS 3.7.1.2 references section 3.5.1.1 for “normal installed monitors.” Confirm this is the correct section. In context, it appears that proposed TS 3.7.1.2 should read, “normal installed monitors in section 3.7.1.1.”