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A SCANA COMPANY

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RC-07-0156

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

ATTN: Mr. R. E. Martin

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING LEAD TEST ASSEMBLIES (TAC NO. MD5699)

References:
1. R. E. Martin (NRC) Letter to J. B. Archie (SCE&G), Virgil C. Summer Nuclear Station - Request for Additional Information Regarding Lead Test Assemblies (TAC NO. MD5699), September 5, 2007
2. J. B. Archie (SCE&G) Letter to Document Control Desk (NRC), Request for Burnup Extension of Exemption Request, May 31, 2007

South Carolina Electric & Gas Company (SCE&G) received an NRC letter dated September 5, 2007 (Reference 1), presenting a request for additional information (RAI) regarding the VCSNS request for burnup extension submitted May 31, 2007 (Reference 2). SCE&G reviewed these questions in consideration of the activities proposed in Reference 2.

SCE&G is providing the attached response to address questions presented in Reference 1.

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NRR

If you have any questions or require additional information, please contact Mr. Bruce Thompson at (803) 931-5042.

Very truly yours,

Jeffrey B. Archie

JT/JBA/mb
Attachment

c: K. B. Marsh
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File (810.32)
DMS (RC-07-0156)

South Carolina Electric & Gas Company (SCE&G)
Virgil C. Summer Nuclear Station (VCSNS)
Response to NRC Request for Additional Information (RAI)
Regarding the Request for Burnup Extension of Exemption Request

RAI Response References:

1. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 23, 1972.

2. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

3. Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003.

4. Dinunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

5. International Commission on Radiation Protection, ICRP-30, "Limits For Intakes of Radionuclides by Workers," 1980.

6. Letter from David J. Modeen (Nuclear Energy Institute) to U. S. NRC, transmitting comments on Draft Regulatory Guide DG-1081, March 31, 2000.

RAI Question:

In your letter dated May 31, 2007, you refer to the VANTAGE+ topical report, concluding that the fuel handling accident (FHA) thyroid doses are not adversely affected by extended burnup. However, the amount of fission gas release (from the fuel pellet) is sensitive to burnup and power history. As such, the fission product gap inventory may be affected by the higher burnup and power history of the LTA. Due to limited fission gas release measurements on high burnup fuel and the lack of an approved fuel performance model up to 75 gigawatt days per metric ton uranium, it is difficult to validate the fission product gap inventory assumed in the FHA assessment. To compensate for this uncertainty in gap inventory, please identify any conservatisms within the existing dose calculation. For example, provide the LTA's peak rod and assembly average power peaking factors during Cycle 18, as compared to the assumed values used in the FHA dose calculation.

Response:

The dose consequences for the Fuel Handling Accident (FHA) are discussed in Section 15.4.5 of the FSAR. This is a standard source term analysis. Limiting results (i.e., 0 – 2 hr thyroid inhalation doses at the exclusion area boundary) are predicted for the FHA within containment. Gap release fractions consistent with Regulatory Guide (RG) 1.25 (Reference 1) and NUREG/CR-5009 (Reference 2) are utilized. A total of 314 pins are assumed to fail. This includes all 264 pins in the dropped assembly and 50 pins in the impacted fuel assembly. This amount of fuel failure is equivalent to 1.19 assemblies.

There are a number of conservatisms within the existing dose calculations which, if credited, would result in a significant reduction in the limiting FHA dose for the Lead Test Assembly (LTA). These include, but are not limited to, conservatisms associated with the following items:

- Pool decontamination factor,
- Assembly relative power,
- Thyroid dose conversion factors,
- Offload time,
- Reactor Building (RB) purge isolation, and
- Mechanical fuel damage due to impact.

Each is described in more detail below.

Pool Decontamination Factor

Consistent with RG 1.25, an overall effective decontamination factor of 100 is utilized in the current analysis to determine the percentage of iodine activity within the fuel rod gap that is released to the RB atmosphere. As described in FSAR Section 15.4.5.1.2.2, this value is a factor of five (5) or more below the expected value. Although not fully credited, this conservatism is recognized in Appendix B to RG 1.195 (Reference 3), which outlines acceptable methodology for evaluating the radiological consequences of a FHA. Provided the depth of the water above the damaged fuel is 23 feet or greater, the accepted decontamination factors for the elemental and organic species of iodine are 400 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine release from the damaged rods is retained by the water). If the RG 1.195 overall effective decontamination factor is credited within the VCSNS FHA analysis, the calculated thyroid dose would decrease by 50%.

Relative Assembly Powers

Due to its high burnup, the LTA's relative power will not approach the 1.7 peaking limit assumed in the FSAR. The highest assembly average power expected during the cycle for the LTA is less than 0.89. The highest rod power is projected to be less than 0.91. Based on assembly average power and accounting for 8% uncertainty, this represents an approximate 35% reduction in the limiting dose (i.e., 42% reduction in 84% of the total fuel rods experiencing damage) for the FHA.

The impacted assembly will also be less than the 1.7 peaking limit assumed in the FSAR. At end of cycle, the highest power assembly is expected to be less than 1.33, with a peak rod less than 1.40. Based on the peak rod power and accounting for 8% uncertainty, this represents an approximate 1.8% reduction in the limiting dose (i.e., 11% reduction in 16% of the total fuel rods experiencing damage) for the FHA.

Therefore, with more appropriate relative assembly powers credited for both the LTA and other potentially impacted assemblies, the limiting dose would decrease by approximately 37%.

Thyroid Dose Conversion Factors

The current dose analysis for the FHA is conservatively based on thyroid dose conversion factors from TID-14844 (Reference 4). Crediting conversion factors from ICRP-30 (Reference 5), which are currently approved for use via issuance of Amendment 162 to the VCSNS Technical Specifications, would result in approximately a 29% reduction in the limiting dose.

Offload Time

The Technical Specifications allow a core offload to begin no sooner than 72 hours after shutdown. In practice, administrative controls ensure that the offload commences no sooner than 72 hours after entry into Mode 3 and ends no sooner than 92 hours after entry into Mode 3.

The current loading pattern calls for the LTA to be loaded in the center of the core and, assuming the use of the normal offload sequence, the LTA is expected to be removed on move 65 of 157. Assuming that the offload starts at the minimum hold time of 72 hours, administrative controls indicate that the LTA would not be moved sooner than 80 hours after entry into Mode 3.

In practice, core offload commences well after 72 hours. For the last 3 outages, it has taken approximately 277, 213, and 188 hours to start core offload. While the time to initiate fuel movement has been getting shorter, it is not expected to approach 144 hours (i.e., 2 times

the 72 hour limit). Based solely on I-131, which is the primary contributor to the thyroid dose and decays with an 8.04-day half-life, an extra 72 hours of decay would result in a 23% reduction in the limiting dose.

Reactor Building Purge Isolation

Following the postulated accident inside the Reactor Building, the radioactivity is assumed to be released to the environment through the Reactor Building Purge System, and no credit is taken for a reduction in the amount of activity released due to filtration or radioactive decay due to holdup in the containment. The offsite radiation doses resulting from the RG 1.25 analysis are within the limits of 10 CFR 100. However, because the estimated thyroid dose is calculated to be higher than the Standard Review Plan 15.7.4 suggested limit (75 Rem), additional instrumentation has been provided to detect the release of radioactivity and to isolate the Reactor Building Purge System. Although not fully safety grade, this equipment is required to be operable during fuel movement and, in the event of a FHA within containment, has been designed to effectively isolate the containment before any radioactivity is released to the environment. For more detail, see FSAR section 15.4.5.1.4.

Mechanical Fuel Damage Due To Impact

The FSAR analysis assumes all rods of the dropped assembly and 50 rods on an impacted assembly fail. This is a very conservative assumption given the broad spectrum of loads (e.g., shipping, thermal, deadweight, LOCA, and seismic loads) considered and the resulting high structural strength of the fuel assembly and other core components. Irradiated fuel assembly drop events (e.g., Ft. Calhoun in 2003, North Anna in 2001, and Haddam Neck in 1986) have also yielded no increase in local area dose rates.

Summary

There are a number of conservatisms within the existing FSAR analysis. Crediting only improvements in the pool decontamination factor, assembly relative powers, and thyroid dose conversion factors, the calculated doses for the FHA would be reduced by approximately 77% (i.e., a factor of ~ 4 reduction).

Gap fractions do increase with increasing burnup. However, available data (per Reference 6) suggests that the increase will be small (i.e., significantly less than a factor of 4 increase) for burnups approaching 75000 MWd/MTU. Given this, there is reasonable assurance that existing conservatisms will more than compensate for the current uncertainty in gap inventory for the LTA.