December 6, 2004

Mr. Michael R. Kansler, President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT APPLICATION FOR ALTERNATE SOURCE TERM (TAC NO. M3351)

Dear Mr. Kansler:

On June 2, 2004, Entergy Nuclear Operations, Inc. (Entergy), submitted an application for a proposed amendment to the Technical Specifications (TSs) for Indian Point Nuclear Generating Unit No. 3 to fully adopt the alternate source term (AST) methodology for design-basis accident dose consequence evaluations in accordance with Section 50.67 of Part 50 of Title 10 of the *Code of Federal Regulations*. Specifically, the amendment would revise the TS Definition regarding dose equivalent iodine and TS Section 5.5.10, “Ventilation Filter Testing Program.” The AST methodology for the fuel-handling accident was previously approved in Amendment No. 215, dated March 17, 2003.

The Nuclear Regulatory Commission (NRC) staff is reviewing the information provided in the June 2 application 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). During a telephone call on December 2, 2004, the Entergy staff indicated that a response to the RAI would be provided within 30 days. In an RAI dated September 30, 2004, the NRC staff previously provided questions regarding containment sump pH and iodine removal coefficients.

Please contact me at (301) 415-1457 if you have any questions on this issue.

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosure: RAI

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING FULL-SCOPE ADOPTION OF ALTERNATE SOURCE TERM

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

In a letter dated June 2, 2004 (ADAMS Accession No. ML041600619), Entergy Nuclear Operations submitted an application for a proposed amendment to the Technical Specifications (TSs) for Indian Point Nuclear Generating Unit No. 3 to fully adopt the alternate source term (AST) methodology for design-basis accident dose consequence evaluations in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.67. The Nuclear Regulatory Commission (NRC) staff has the following questions regarding the information provided.

Attachment 1 - Proposed Changes

1. The proposed re-definition for dose equivalent iodine isotope 131 (^{131}I) allows the use of the committed effective dose equivalent (CEDE) dose conversion factors for either the submersion, inhalation and ingestion pathways. The Definition must be modified to indicate that it is only the inhalation pathway CEDE dose conversion factors.

2. What is the basis for including ^{130}I in the calculation of dose equivalent ^{131}I?

3. It has been proposed that the testing requirements for the fuel storage building emergency ventilation system be deleted. It appears that the basis for its removal is the fact that it has been determined that a fuel-handling accident occurring within containment results in acceptable offsite and control room operator doses without the assumption of containment integrity and without credit for filtration. This filtration system has been utilized to reduce the release of effluents during refueling operations. Section II.D of Appendix I to 10 CFR Part 50 requires licensees to include in their radwaste systems all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. Your Appendix I analysis assumed filtration of the effluents during fuel-handling operations. Provide your analysis which demonstrates that removal of the fuel storage building emergency ventilation system is in compliance with Section II.D of Appendix I.

4. If containment integrity is not established for a fuel-handling accident and the fuel storage building emergency ventilation system is not operating, explain how the requirements of General Design Criteria (GDC) 60, 61 and 64 are met during these fuel-handling operations.

Attachment I - Technical Analysis

5. Although you have supplied an analysis assuming removal of the spray additive tank, the NRC staff's assessment of this amendment will not include a review of that analysis unless you indicate that you are seeking approval of the removal of the spray additive tank.

6. It was indicated that for each of the accident scenarios two different control room heating, ventilation, and air conditioning (HVAC) configurations were analyzed. If the intent is to have the option of selecting either configuration in the event of a radiological accident, then it is appropriate to assess both configurations. However, if it intended that there will only be one configuration, which will be the method of operation for the control room HVAC, then only that configuration will be assessed by the NRC staff. What is the intended mode(s) of operation of the control room HVAC in the event of a radiological accident?

7. Will the analysis be amended and submitted to the NRC staff for review and approval if it is determined that the test results of the containment spray system pump show that the penalty applied to the containment spray system flow rate did not provide adequate margin?

8. The table summarizing the dose limit for the various accidents had incorrect limits for the gas decay tank rupture, the volume control tank rupture and the holdup tank rupture. For AST, the limit should be from 10 CFR Part 20 (i.e., 100 mrem total effective dose equivalent (TEDE)). This necessitates that these three accidents need to be re-analyzed to meet the 100 mrem TEDE acceptance criteria or the proposed switch to AST for these three accidents be withdrawn.

Attachment II - Program 5.5.10

9. Items a and b of the Program indicate that the in-place acceptance criteria is based upon a penetration of no more than 1%. Has your analyses included a reduction in the effectiveness of the high efficiency particulate filters and the charcoal adsorber to account for this 1% penetration?

10. Explain why the 1-inch bed of the control room HVAC system is only required to remove 93% of the methyl iodine at a face velocity of 50 feet per minute (ft/min) but must remove 95.5% at a face velocity of 78 ft/min.

Attachment III - Radiological Consequences of Accidents

11. Control room operator doses are provided. For which control room emergency ventilation system operating mode do these doses pertain?

Loss-of-Coolant Accident (LOCA)

12. At what time following the LOCA was the decontamination factor (DF) of 1000 achieved for particulate? What isotopes was the DF based upon? Provide your calculation that determined when the DF was achieved.

13. What is the basis for assuming that the airborne fraction of the leakage from the reactor coolant pump is 10%?

14. What is the basis for assuming that the airborne fraction of emergency core cooling system (ECCS) leakage is 2.7% starting at 6.5 hours after the accident and not a minimum of 10%?

15. What is the basis for the assumption that there is no sump leakage or reactor coolant pump seal leak-off line leakage between 4 and 6.5 hours?

Locked Reactor Coolant Pump Rotor

16. The analyses of the consequences to control room operators should reflect the inleakage characteristics of the control room envelope for the various modes of operation during a radiological accident. Provide the inleakage characteristics of the control room envelope when the normal control room ventilation system is operating and during the time that control room operators are taking the manual actions to place the control room emergency filtration system into operation.

Rod Ejection

17. What is the basis for assuming that it will only take 2 hours to stop steam releases from the steam generators and initiate residual heat removal (RHR) when it takes considerably longer to initiate RHR for other accidents?

18. What is the control room envelope inleakage rate during normal operation for this accident?

Small-Break LOCA

19. What is the basis for assuming that it will only take 2 hours to stop steam releases from the steam generators and initiate RHR when it takes considerably longer to initiate RHR for other accidents?

Fuel-Handling Accident

20. In Section 11.1.4, it is stated that credit has not been taken for filtration or containment isolation and that the IP3 analysis supports refueling operations with the equipment hatch and personnel air lock remaining open. The acceptability of these apertures during fuel-handling operations is not limited to obtaining acceptable control room and offsite dose consequences. It is also necessary to demonstrate that the facility remains able to meet GDC 60 and 64. It is also necessary that the removal of such equipment during fuel-handling operations meet the criterion of Section II.D of Appendix I to 10 CFR Part 50. Provide additional justification for the equipment hatch and personnel air lock remaining open during fuel-handling operations.

Conclusions

21. Attachment III does not support the complete removal of all filters from the fan coolers if trisodium phosphate baskets are installed as is stated in this section. It was previously stated in the submittal that an analysis justifying removal of the spray additive was provided for information purposes only and was not part of this amendment request. Since there was no request for the NRC staff to review the analysis, the acceptability of the use of trisodium phosphate baskets in lieu of containment fan coolers cannot be presumed. This section needs to be modified to clarify that this submittal is not a justification for the use of the trisodium phosphate baskets.