May 24, 2001

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing and Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 — REQUEST FOR ADDITIONAL INFORMATION RE: PROPOSED LICENSE AMENDMENT REQUEST N0. 262, REVISION 0, ALTERNATIVE SOURCE TERM AND CONTROL ROOM EMERGENCY VENTILATION SYSTEM (TAC NO. MB0241)

Dear Mr. Young:

By letter dated October 3, 2000 (3F1000-08), you submitted an amendment application to revise the Crystal River Unit 3 (CR-3) Improved Technical Specification (ITS) 3.7.12, "Control Room Emergency Ventilation System (CREVS)," ITS 5.6.2.12, "Ventilation Filter Testing Program (VFTP)," ITS 3.3.165, "Control Room Isolation - High Radiation," and ITS 3.7.18, "Control Complex Cooling System." The staff is currently reviewing your request, and the reviewers have determined that additional information is needed. The questions are listed below, and questions 1 through 6 and 8 through 11 were previously discussed with the staff in a May 3, 2001, telephone call.

The following additional information is requested to enable the staff to determine if your analysis demonstrates compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

1. There appears to be a discrepancy between the atmospheric dispersion values (x/Q) used for the main steamline break, steam generator tube rupture, and waste gas decay tank analyses and those stated in Table 14-23 of the CR-3 Final Safety Analysis Report (FSAR). In your letter of November 7, 1997, the response to staff question #7 states that the values in FSAR Table 14-23 are to be used for assessing these events. While the values used in the analysis are more conservative for the 0-to-8-hour period than the values provided in Table 14-23, they are not consistent with the design basis. Please identify whether or not it is your intent to revise the current design basis to use the more conservative values.

2. Your analyses have used a single set of x/Q values for assessing control room dose from all analyzed accidents. Please explain why these values are bounding for each of the combinations of release points and control room intakes, given differences in the intervening distances and in wind direction frequency.

3. In several locations, you have stated that an analysis assumption or input was based on Draft Regulatory Guide (DG)-1081 or on NUREG-1465. In some of these occurrences, you have stated that the DG-1081 values are more conservative than those in Regulatory Guide (RG) 1.183. Except where explicitly identified in this Request for

Additional Information (RAI), the staff agrees with your position that the current analyses are adequate for this amendment request and need not be re-analyzed. However, the staff does not believe that it is appropriate to reference DG-1081 and NUREG-1465 in the CR-3 design basis. DG-1081 is a draft document that was revised in response to public comments. The official agency guidance is provided in RG 1.183. While NUREG-1465 was the source of some of the guidance provided in RG 1.183, the staff has not endorsed NUREG-1465 as an acceptable approach to demonstrating compliance to 10 CFR 50.67. Please provide a commitment that the applicable assumptions of RG 1.183 will constitute the CR-3 design basis and that future revisions to these analyses will utilize the applicable guidance in RG 1.183 or acceptable alternatives thereto.

4. In Appendix I to your application, you have described your analysis of the consequences of emergency core cooling system (ECCS) leakage, assuming, in part, a 50-gallon per minute release lasting 30 minutes from passive failure of ECCS piping at 24-hours postaccident. This analysis requirement of DG-1081 was deleted from RG 1.183. Your assumption does result in separate maximum 2-hour doses for each release path. Instead of conservatively summing the projected doses, you have considered only the ECCS leakage that occurs in the 0.8 to 2.8 hour maximum dose period for containment leakage dose. You state that this is appropriate since “It can be assumed that the population at the exclusion area boundary (EAB) would have been evacuated by 24 hours . . . .” The staff cannot accept this rationale in a design basis accident assessment. The language of the dose criterion addresses the maximum exposure to any individual at any point on the boundary of the EAB for any 2-hour period. There is a presumption that an individual is present for the entire exposure period. The decision to actually implement the evacuation rests with the elected state and local officials. While it may be reasonable to presume that these officials will respond appropriately in emergency preparedness space, CR-3 cannot control whether or not an evacuation will occur. Please revise your analysis and submittal or provide additional justification for your position.

5. The control-rod-ejection accident used in your analysis was based on the core average source term used in RADTRAD. Since the fuel clad breach is limited to only 14% of the core, the source term needs to be adjusted for the radial peaking factor. See Regulatory Position 3.1 of RG 1.183. You used a peaking factor of 1.8 in the fuel-handling accident (FHA) analysis. The staff analysis assuming a radial peaking factor of 1.8 projected doses within acceptance criteria, but with a control dose greater than that for the loss-of-coolant accident (LOCA). Please revise your analysis and submittal or provide additional justification for your position.

6. For the iodine spiking releases shown in the tables in Attachment 2 to Appendix E and in Attachment 2 to Appendix L, your submittal explains that they were generated by a thermohydraulic code. Page 8 of Appendix B does explain some of the basis of this determination for the letdown break. Please provide a brief description of the calculational algorithm used by this code. Also provide the volume or mass of primary and secondary releases for the four time periods. These data are needed to assess the acceptability of your assumed releases.

7. In your application, you propose a breach margin of 50 square inches. The derivation of this breach size is provided on page 7 of Appendix B as being based on the margin between the 4.29 roentgen equivalent man (rem) LOCA result and the 5-rem dose criterion. However, page 4 of the same appendix indicates that the dose due to an FHA is 4.43 rem. This information is also tabulated in the control room habitability report, which treats the LOCA as the maximum hypothetical accident (MHA). Please explain why the FHA is not the MHA with regard to control room habitability since the projected dose is greater and why the FHA is not being used as the basis for the breach margin. Please confirm that the results of the control-rod-ejection accident which you are re-assessing in response to the earlier RAI does not establish the control room envelope (CRE) as the MHA.

8. CR-3, in their submittal dated October 3, 2000, provided no basis for the acceptability of equating their unfiltered in-leakages inside the control complex habitability envelope (CCHE) with a conversion factor of "8.02 cfm for one square inch breach." Also, CR-3 provided no information regarding the margin of errors associated with this approximation. In addition, CR-3 did not provide an explanation for why the conversion factor of "8.02 cfm [cubic feet per minute] for one square inch breach" is valid for various hole sizes and/or multiple holes, which add up to the aggregate in-leakage as specified in the proposed TS Bases. Appendix B, page 7 of 9 to your submittal, refers to 8.02 cfm for 1 square inch breach, based on 0.2-inch water gauge (WG) pressure differential between CRE and auxiliary building. In order to assess compliance with the requirements of 10 CFR 50, Appendix A, General Design Criterion 19 and/or 10 CFR 50.67, please provide the following:

- CREVS flow diagram showing flow and pressure data serving the CCHE.

- Bases and assumptions made in the calculation of the conversion factor of 8.02 cfm for 1 square inch breach, and the margin of errors associated with it. Also, provide Calculation Number M97-0137, Revision 4.

- A detailed explanation stating why the conversion factor of 8.02 cfm for 1 square inch breach is valid for the various hole sizes and/or multiple holes which add up to an aggregate hole size of 50 square inches for the aggregate leak rate of 400 cfm as specified in the proposed ITS Bases.

- Please number the separate locations where the actual pressure differentials were measured during the tracer gas testing. Also, describe where these measurements were taken with respect to the CCHE and state if they were of the same value.

- Basis for the extrapolation methodology used to determine the unfiltered inleakages at the worst-case pressure differential of 0.2-inch WG from the actual test results at lower pressure differentials (i.e., 513 cfm at 0.2-inch WG corresponds to the actual measured leak rate of 443 ± 20 at 0.171-inch WG and 503 cfm at 0.2-inch WG corresponds to the actual measured leak rate of 450 ± 13 cfm at 0.176-inch WG).

9. Appendix A, Section III.2, page 6 to your submittal states that “The maximum differential across the CCHE occurs when there is no loss of offsite power. In this case, the auxiliary building exhaust fans would remain on and the supply fans would trip on high radiation. This results in a negative pressure in the auxiliary building and hence a higher differential pressure across the CCHE of 0.2-inch WG.” Does this consider the worst-case alignment of the other adjacent ventilation systems? Provide the bases for why the pressure differential of 0.2-inch WG is the maximum expected pressure differential during all accident conditions.

10. Appendix A, Section III.2, page 7 to your submittal states that “The 1999 test results were essentially equal to the 1997 results, thus demonstrating that the CCHE boundary is not degrading with the time.” Provide your reasoning why two tracer tests conducted during a 2-year period (1997 through 1999) demonstrate that the CCHE will not degrade over a longer period.

11. The proposed ITS 3.7.12, “CONDITION A,” allows the restoring of the CCHE boundary, due to a breach or breaches in excess of the limit, within “COMPLETION TIME” of “24 hours.” This reflects the concept of Technical Specification Task Force (TSTF)-287, but not in its format or wording. In order for the Nuclear Regulatory Commission (NRC) staff to find the requested “24-hour completion time” acceptable, the licensee needs to provide a formal submittal request in accordance with the TSTF-287 requirements, which has been generically approved by the NRC staff.

For the staff to complete its review schedule, your response is appreciated prior to June 15, 2001. This date was mutually agreed on in a telephone conversation with CR-3 personnel on May 4, 2001. If circumstances result in the need to revise the target date, please call me at the earliest opportunity.

Sincerely,

/RA/

John M. Goshen, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

cc: See next page

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11. The proposed ITS 3.7.12, “CONDITION A,” allows the restoring of the CCHE boundary, due to a breach or breaches in excess of the limit, within “COMPLETION TIME” of “24 hours.” This reflects the concept of Technical Specification Task Force (TSTF)-287, but not in its format or wording. In order for the Nuclear Regulatory Commission (NRC) staff to find the requested “24-hour completion time” acceptable, the licensee needs to provide a formal submittal request in accordance with the TSTF-287 requirements, which has been generically approved by the NRC staff.

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John M. Goshen, Project Manager, Section 2
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