Entergy

Entergy Operations, Inc.
River Bend Station
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RBG - 46053

December 20, 2002

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

SUBJECT: River Bend Station, Unit 1
Docket No. 50-458
Supplement 2 to Amendment Request
Full-Scope Application of NUREG-1465 Alternative Source Term Insights TAC No. MB5021.

REFERENCES:
1. Letter RBG-45930 dated April 24, 2002 from Entergy to USNRC, "License Amendment Request, Full Scope Application of NUREG-1465 Alternative Source Term Insights"
2. Letter RBG-45989 dated July 18, 2002 from Entergy to USNRC, "Supplement to License Amendment Request, Full Scope Application of NUREG-1465 Alternative Source Term Insights."

Dear Sir or Madam:

By Reference 1, Entergy Operations, Inc. (Entergy) proposed a change to the River Bend Station, Unit 1 (RBS) Operating License and Technical Specifications (TSs) associated with a full scope application of NUREG-1465, Alternative Source Terms. Reference 2 provided supplemental information on the details of the analysis.

On July 17, 2002, Entergy received a facsimile transmission of eight additional draft questions being considered by members of your staff. Additionally, the USNRC and Entergy held a public meeting on October 8, 2002 to discuss the application, the draft questions received to date and other NRC staff concerns. At the meeting, the NRC staff requested Entergy to address one additional question regarding Control Room habitability. Entergy's response to the nine questions is contained in Attachment 1.

There are no technical changes proposed to the alternate source term analyses. However, Entergy is withdrawing a portion of the proposed TS changes. Revised TS pages reflecting these changes will be submitted under separate letter. The original no significant hazards considerations included in Reference 1 is not affected by any information contained in the supplemental letter. Entergy has also reviewed the Federal Register Notice published on June 11, 2002 (Volume 67, Number 112, pages 40021 and 40022) for the amendment request. Entergy believes the description of the proposed amendment in the FR notice and the evaluation of the finding that no significant hazards considerations were involved are still valid as published.

A001
A003

There are 2 new commitments contained in this letter. These commitments are listed in Attachment 2.

If you have any questions or require additional information, please contact Greg Norris at 225-336-6391.

I declare under penalty of perjury that the foregoing is true and correct. Executed December 20, 2002.

Sincerely,

William R. Brian
Director, Engineering

WRB/rwb

Attachments:
1. Response to Request For Additional Information
2. List of Regulatory Commitments

cc: U. S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
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U.S. Nuclear Regulatory Commission
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Mr. Prosanta Chowdhury
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Attachment 1

To

RBG- 46053

Response to Request for Additional Information

Response to Request for Additional Information Related to Full Scope Application of Alternate Source Terms

Question:

1. In your radiological dose calculations, you assumed and requested to have manual initiation (instead of current automatic initiation) of the Standby Gas Treatment System, the Main Control Room Emergency Fresh Air Emergency Filtration System, and the Main Steam Positive Leakage Control System within 20 minutes from the initiation of the postulated design basis accidents (DBAs). In Attachment 1, you further stated that "operator notification to take the appropriate manual actions in accordance with plant procedures is assured." Which plant operating procedures and/or emergency operating procedures will require these manual initiations? In Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," the staff stated that while compliance with the General Design Criteria of Appendix A to 10CFR50 for meeting the relevant dose criteria is essential, modifications proposed for the facility by the licensee generally should not create a need for compensatory programmatic activities such as reliance on manual operator actions.

   You further stated in your transmittal letter dated April 24, 2002, that you will be implementing the amendment within 60 days from the NRC approval. Will you be revising the appropriate plant operating procedures to incorporate the above manual initiation requirements within this time period?

Response:

No physical plant modifications were proposed by the application and were not planned based on the full scope application of Alternate Source Terms (AST) for River Bend Station (RBS). The Main Steam-Positive Leakage Control System is currently a manually actuated system consistent with the current plant design and licensing bases.

Automatic isolation of the secondary containment and corresponding initiation of the Standby Gas Treatment (SGTS) system was assumed in the Loss of Coolant Accident (LOCA) dose analysis currently summarized in the updated Final Safety Analysis Report (FSAR) Chapter 15. Automatic initiation of CRFA due to either a high radiation or LOCA signal is also currently assumed in all of the major accidents summarized in updated FSAR Chapter 15. As such, the instrumentation providing the automatic initiation signals for both SGTS and CRFA currently meet 10CFR50.36.c.2.ii(C) Criterion 3 and have Limiting Conditions for Operations (LCOs) established in the Technical Specifications (TS).

The new AST analyses did not assume any automatic initiation of these systems. Thus, even though Entergy had no plans at the time to modify the plant, Entergy did propose to revise the corresponding TS LCOs for these automatic features since only the manual function met Criterion 3.

Entergy has reassessed the proposed changes and has decided to withdraw them at this time. Since Entergy does not plan to physically modify or remove the automatic features as installed in the plant, the benefits of the TS changes are minimal. In addition, Entergy believes that other generic TS change initiatives may provide commensurate benefits. Entergy will not revise the AST analysis for this application. The analysis assumption that the system actuations are delayed for 20 minutes due to manual rather than automatic actuation is a conservative assumption that continues to support the remaining proposed changes.

Specifically, Entergy is withdrawing the following TS changes proposed by Reference 1:

- all changes to LCO 3.3.6.2, Secondary Containment and Fuel Building Isolation Instrumentation, except for the deletion of Fuel Building isolation instrumentation,
- all changes to LCO 3.3.7.1, Control Room Fresh Air System Instrumentation, except for Note (b) of Table 3.3.7.1-1.

Revised pages reflecting these changes will be submitted under separate letter.

Entergy has also reviewed its no significant hazards considerations provided in Reference 1 and reviewed the Federal Register Notice published on June 11, 2002 (Volume 67, Number 112, pages 40021 and 40022) for the amendment request. Entergy believes the description of the proposed amendment in the FR notice and the evaluation of the finding that no significant hazards considerations were involved are still valid as published.

Question:

2. In Attachment 4, you listed one regulatory commitment to add the main control room air intake monitors to Technical Requirements Manual, TLCO-3.3.7.1 to provide indication to operators so that they can select the more favorable intake. Provide a copy of TLCO-3.3.7.1. Does this change require a plant modification? Are these radiation monitors safety related? Will these changes be included in plant operator training?

Response:

As discussed in response to question 1 above, Entergy is withdrawing the request to delete the TS requirements for the automatic function of the main control room (MCR) local main air intake (MAI) radiation monitors. Therefore, we no longer need to revise TLCO-3.3.7.1 and the commitment is withdrawn.

The existing radiation monitors on both the local air intake and the remote air intake (RAI) are safety-related. During normal operations the plant is typically aligned to the MAI. Main Control Room Panel H13-P863 has annunciators which alarm and automatically signals initiation of the CRFA System on a high radiation signal. Each radiation monitor has an individual alarm for a total of 4 annunciators. Plant Alarm Response Procedures (ARP-863-74, Revision 11) require that if an alarm occurs due to high or increasing radiation signal, then Operations must shift to the RAI. Further, the ARP directs the operator to monitor the RAI radiation monitors and take further actions as appropriate.

Question:

3. Provide a general building layout figure similar to that provided in Figure 1 of Attachment 6 designating the following:
   - Drywell
   - Containment
   - Annulus
   - Annulus Bypass Leakage Paths
   - Fuel Building
   - Secondary Containment Envelope
   - Secondary Containment Envelope Leakage Path
   - Containment/Fuel Building Personnel Airlock
   - Inclined Fuel Transfer System Drain Lines
   - Standby Gas Treatment System
   - Auxiliary Building
   - Control Building
   - Control Room
   - Control Room Main Air Intake
   - Control Room Remote Air Intake
   - All Source Term Release Points (LOCA, Fuel Handling Accident, Control Rod Drop Accident, and Main Steam Line Break Accident)

Response:

Entergy presented a General Building Layout figure in Reference 1 and at the October 8 meeting. However, the layout does not show all requested information. Additional information where more detailed drawings can be found are provided or referenced below.

- **Drywell:** See USAR Figure 1.2-7
- **Containment:** See USAR Figure 1.2-7
- **Annulus:** See USAR Figure 1.2-7
- **Annulus Bypass Leakage Paths:** RBS Technical Requirements (TRM) Section 3.6.1.1 contains the list of Annulus Bypass leakage paths. These paths are listed in Table 1 below.

Table 1
Annulus Bypass Leakage Paths

Penetration	Valve/ Component Number
JRB-CRD	CRD HATCH
KJB-Z601B	SSR-SOV131 SSR-SOV130
KJB-Z601E	SSR-SOV133 SSR-SOV134
KJB-Z601F	SSR-V706 SSR-SOV140
JRB-DRA1	AB Airlock
KJB-Z31	HVR-AOV165 (CPP-SOV140) HVR-AOV123

- **Fuel Building:** See USAR Figures 1.2-6 and 1.2-20 through 1.2-23.
- **Secondary Containment Envelope:** Consists of the Auxiliary Building and the Shield Building Annulus. See USAR Figures 1.2-6 and 1.2-7.
- **Secondary Containment Envelope Leakage Paths:** The Secondary Containment Leakage Paths are listed in Table 2 below.

Table 2
Secondary Containment Bypass Leakage Paths

CONT. PEN. NUMBER	SYSTEM NAME	FLUID	LINE SIZE (INCH)	LEAKAGE DESTIN-ATION
1KJB*Z3A	FEEDWATER LINE	FDW FDW FDW	20 20 20	TURBINE
1KJB*Z3B	FEEDWATER LINE	FDW FDW FDW	20 20 20	TURBINE
1KJB*Z41	FIRE PROTECTION HEADER (14)	WATER WATER	6 6	YARD TO FIRE PUMP HOUSE
1KJB*Z44	SERVICE AIR SUPPLY TO CONTAINMENT AND DRYWELL (14)	AIR AIR	4 4	TUNNELS & TURBINE BLDG.
1KJB*Z46	INSTRUMENT AIR SUPPLY TO CONTAINMENT AND DRYWELL (14)	AIR AIR	3 3	TUNNELS & TURBINE BLDG.
1KJB*Z131	VENTILATION CHILLED WATER RETURN	WATER WATER WATER	8 8 3/4	TURBINE BUILDING UNIT COOLERS
1KJB*Z132	VENTILATION CHILLED WATER SUPPLY	WATER WATER	8 8	TURBINE BUILDING UNIT COOLERS
1KJB*Z134	CONDENSATE MAKEUP SUPPLY	WATER WATER	4 4	TURBINE
1KJB*Z26 (NOTE)	FUEL POOL COOLING AND CLEANUP DISCHARGE	WATER WATER	12 12	FUEL BUILDING CLOSED SYSTEM
1KJB*Z27 (NOTE)	FUEL POOL COOLING AND CLEANUP SUCTION LINE	WATER WATER WATER	12 12 3/4	FUEL BUILDING CLOSED SYSTEM
1KJB*Z28 (NOTE)	FUEL POOL PURIFICATION SUCTION	WATER WATER WATER	8 8 3/4	FUEL BUILDING CLOSED SYSTEM
1KJB*Z29 (NOTE)	CRD HYDRAULIC SYSTEM SUPPLY	WATER WATER	2 2	FB, TUNNEL, THEN YARD (CST)
IFTS	INCLINED FUEL TRANSFER SYSTEM	WATER	4	FB IFTS DRAIN TANK
AIRLOCK	AIRLOCK	-	-	FB

Note: KJB-Z26 through –Z29 are not currently considered in the SCB summation.

- **Containment/Fuel Building Personnel Airlock:** The Fuel Building Personnel Airlock (JRB-DRA2) is located on the 113' elevation. See USAR Figure 1.2-6.

- **Inclined Fuel Transfer System Drain Lines:** See USAR Figure 9.1-20 and 9.1-22. Also, a significant amount of information on the IFTS system was provided to the NRC via RBS Letter RBG-45202, dated December 20, 1999 (Submittal of LAR 99-30, "IFTS Blind Flange").

- **Standby Gas Treatment System:** SGTS is discussed in USAR Section 6.2.3.2.1. The SGTS P&ID is located on USAR Figure 6.2-58. USAR Figure 12.3-12 contains a schematic of the filter trains themselves.

- **Auxiliary Building:** See USAR Figures 1.2-6 and 1.2-13 through 1.2-19.

- **Control Building:** See USAR Figures 1.2-24 through 1.2-27.

- **Control Room:** See USAR Figures 1.2-24 through 1.2-27.

- **Control Room Main Air Intake**: The location of the MAI can be found on Figure 1 of Attachment 6 of the original submittal.

- **Control Room Remote Air Intake:** The location of the MAI can be found on Figure 1 of Attachment 6 of the original submittal.

- **All Source Term Release Points (LOCA, Fuel Handling Accident, Control Rod Drop Accident, and Main Steam Line Break Accident)**

  - **LOCA:**
    - Containment and Secondary releases are based on the Standby Gas Treatment release point (main plant stack).
    - Main Steam Isolation Valve and Secondary Containment Bypass leakage terms assume a Turbine Building release point.
    - Engineered Safety Features liquid leakage releases are based on the Standby Gas Treatment release point (main plant stack).

  - **Control Rod Drop Accident:** The releases for both scenarios are based on a turbine building release point.

  - **Main Steam Line Break:** The MSLB assumes a release from the main steam tunnel blowout panel.

  - **Fuel Handling Accident:** The FHA assumes a release from the primary containment building, however, the values are consistent with those used in the current FHA analyses (See Amendments 25, 85, and 114), i.e., the values are based on the Murphy-Campe methodology. Confirmatory calculations were performed which demonstrate that the Murphy-Campe values are conservative.

Question:

4. In page 2 of 17 to Attachment 1, you stated that under a design basis LOCA, all potential leakage paths to the fuel building were assumed to be released directly to the environment. State the leakage paths, the leakage rates, and leakage duration.

Response:

Currently there are two containment penetrations which are identified in the secondary containment bypass (SCB) summation. The first is the containment personnel airlock (PAL) located on the 113' elevation of the fuel building. This airlock has an administrative limit of 60 sccm per RBS procedure ADM-0050, Revision 11, however, the requested SCB limit assumed an administrative limit of 90 sccm. The second SCB leakage path to the fuel building is the Inclined Fuel Transfer System (IFTS) drain line. This leakage path is only of concern with the IFTS blind flange removed (See RBS TS Amendments 116 and 117). That line has an administrative limit of 1000 sccm. The blind flange itself has an administrative limit of 20 sccm.

USAR Table 6.2-40 contains the list of penetrations through the primary containment building. During preparation of AST review of this Table identified four additional penetrations which lead to the fuel building. Three of the penetrations support the fuel pool cooling and cleaning system. The remaining penetration supports the control rod drive (CRD) system. These four penetrations were also considered in the proposed SCB limit of 580,000 cc/hr (@ P_a).

The paths summarized in Table 3 below were considered as part of the SCB leakage term. As such they were assumed to leak to the environment from the onset of fuel damage for the duration of the event. Note that the RBS model did not have a level of detail to consider individual leakage paths, i.e., SCB leakage was treated as one term leaking at the Technical Specification allowable value for the duration of the event as prescribed by Regulatory Guide 1.183. Individual terms were not individually addressed.

Table 3
Containment Penetrations to Fuel Building

Penetration	Valve/Component Number	Desired Admin. Limit (sccm)	Notes
KJB-Z26	SFC-MOV119	2,400	Not currently in SCB summation, but penetration leads to Fuel Building.
	SFC-V101	2,400	
KJB-Z27	SFC-MOV120 SFC-V350	2,400 200	Not currently in SCB summation, but penetration leads to Fuel Building.
	SFC-MOV122	2,400	
KJB-Z28	SFC-MOV139 SFC-V351	1,600 200	Not currently in SCB summation, but penetration leads to Fuel Building
	SFC-MOV121	1,600	
KJB-Z29	C11-MOVF083	400	Not currently in SCB summation, but penetration leads to Fuel Building.
	C11-VF122	400	
IFTS	F42-MOVF003	1,000	Only a contributor when the IFTS blind flange is removed.
Fuel Bld Airlock	JRB-DRA2	90	Transferred from Annulus Bypass summation.

Question:

5. Reference how the post-LOCA suppression pool pH evaluation in Attachment 5 is used in the radiological consequence analysis of the postulated design basis LOCA in Attachment 7, "LOCA Dose Analysis Summary."

Response:

Regulatory Guide 1.183, Appendix A, Section 2 states that "If the sump or suppression pool pH is controlled at values of 7.0 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodine (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodine." The suppression pool pH analysis, as summarized in Attachment 5 of the RBS submittal, demonstrates that the pH would remain above 7.0 (after the initial transient). Since the pH remains above 7.0, the chemical fractions of radioiodine prescribed by the RG were utilized in the RBS dose analysis.

Question:

6. State the decay time of 24 hours in Attachment 8 and in its Tables 2 and 3. Provide the noble gases and iodine activities in the fuel rod gap prior to fuel movement that is available for release to the water surrounding the failed fuel assembly. Also, provide the amounts of fission product activities (in curies) released to the environment following the postulated fuel handling accident.

Response:

The decay time used in both the AST Fuel Handling Accident (FHA) and Light Load Accident (LLA) analyses was 24 hours. The activities released to the environment following a FHA are presented in Table 4 below.

Table 4
FHA Activities

Isotope	Core Concentration t = 0 (Ci/MWt)	Gap Activity Released to Water(1) t = 0 (Curies)	Integral Activity Released to Environment(2) t = 744 hours
I-131	2.6E+04	4.2E+04	1.9E+02
I-132	3.8E+04	3.9E+04	1.0E-01
I-133	5.5E+04	5.5E+04	1.2E+02
I-134	6.1E+04	6.1E+04	8.5E-07
I-135	5.2E+04	5.2E+04	1.9E+01
I-136	N/A	N/A	N/A
Kr-83m	N/A	N/A	N/A
Kr-85	3.0E+02	6.1E+02	6.1E+02
Kr-85m	6.7E+03	6.8E+03	1.4E+02
Kr-87	1.3E+04	1.3E+04	1.6E-02
Kr-88	1.8E+04	1.8E+04	4.1E+01
Kr-89	N/A	N/A	N/A
Xe-131m	N/A	N/A	N/A
Xe-133m	N/A	N/A	N/A
Xe-133	5.5E+04	5 6E+04	4.9E+04
Xe-135m	N/A	N/A	N/A
Xe-135	7.1E+03	7.2E+03	1.1E+03
Xe-137	N/A	N/A	N/A
Xe-138	N/A	N/A	N/A

Note 1: These activities represent the values immediately following shutdown (t = 0 hours). The activity is decayed using RADTRAD.

Note 2: This column was obtained from the RADTRAD output file and represents the total activity released over the 30 day event (plus 24 hours which is the earliest time an FHA could occur).

Question:

7. In Attachment 8, "Fuel Handling Accident and Light Load Drop Dose Summary," reference the following previous River Bend license amendments issued.

    - License Amendment No. 85, "Primary Containment Airlocks," issued on January 11, 1996.
    - License Amendment No. 110, "Revision to Fuel Handling Accident Dose Calculation," issued on March 2, 2000.
    - License Amendment No. 113, "Removal of Fuel Building and Fuel Building Ventilation System from Secondary Containment System Boundary," issued on September 22, 2000, and
    - License Amendment No. 114, "Operational Conditions for Handling Irradiated Fuel in the Primary Containment."

    Provide the chronological changes in the major parameters used in the postulated fuel handling accident dose calculations.

Response:

RBS has pursued a number of Technical Specification amendments directly and indirectly related to the FHA doses analyses. Only one FHA scenario was analyzed to support the initial licensing of the plant. Specifically, drop of irradiated fuel was postulated to occur in the fuel building 24 hours post-shutdown. Credit was taken for the CRFA and fuel building ventilation ESF charcoal filters in that analysis. The methodology used in that analysis was consistent with Regulatory Guide 1.25 guidance. A separate analysis was not performed for the containment building since containment integrity was required during shutdown so the fuel building was the bounding scenario (consistent with SRP guidance).

A summary of the licensing amendments concerning the FHA dose analyses are as follows:

- Amendment 35 was issued on March 3, 1989. This amendment allowed opening of up to 12 vent and drain lines in the primary containment during Local Leakage Rate Testing (LLRT) 70 hours post-shutdown. To support this analysis an FHA in the primary containment was analyzed. The leakage rate assumed was L_a (0.26 %/day) plus an additional 70.2 cfm through the open vent and drain lines.

- Amendment 85 was issued on January 11, 1996. This amendment allowed the containment personnel airlocks (PAL) to be opened except during movement of "recently irradiated" fuel (i.e., fuel which was part of a critical reactor core within the previous 11 days). An analysis was performed assuming a drop of irradiated fuel in containment 11 days after shutdown. No credit was taken for secondary containment, the fuel building, or their associated ESF charcoal filtration systems.

- Amendment 110 was issued on March 2, 2000. This amendment approved revision to the three FHA analyses (24 hours in the fuel building, 70 hours in containment during LLRT testing, and 11 days in containment with the PAL open). Revision to the analyses was required for several reasons. First, the Radial Peaking Factor (RPF) was increased from 1.5 (assumed per RG 1.25) to 1.65. Also, the gap fraction for I-131 was increased from 0.10 (assumed per RG 1.25) to 0.12 to account for the potential impact of extended burnup fuels (per NUREG/CR-5009 a value of 0.12 bounds fuel burnups of up to 60 GWd/MT). Also, the release rate for the 11 day analysis was revised to ensure that the RG 1.25 2-hour release duration was met. Amendment 110 conservatively assumed a power level consistent with the RBS power uprate (3,039x1.02 = 3100 MWt).

- Amendment 113 was approved on September 22, 2000. This amendment removed all requirements concerning the fuel building during normal operation and shutdown, except during movement of "recently irradiated fuel." No revision to the FHA dose analysis was required as the 11 day containment analysis bounded movement of irradiated fuel in the fuel building. Specifically, since containment integrity was not credited the 2 hour release duration assumed in the PAL analysis (per RG 1.25) was appropriate for the fuel building with the cask handling doors open.

- Amendment 114 approved a 5% increase in core rated thermal power, i.e., and increase from 2,894 MWt to 3,039 MWt. Since the uprated power level was accounted for in the Amendment 110 submittal, no additional changes to the analyses were required.

- Amendment 119 was approved on September 14, 2001. This amendment utilized TS Task Force Traveler (TSTF) No. 51, Revision 2, to justify deletion of the requirement for containment integrity in Mode 5 (other than during movement of "recently irradiated" fuel). As discussed above the 11 day analysis utilized RG 1.25 assumptions and did not credit primary or secondary containment, thus, a revision to the FHA dose analyses was not required to support this amendment.

Table 5
RBS FHA Analyses Licensing Changes

Amendment	Analyses Affected			Summary
	24 hour – FB	70 hour – Cont.	11 day – Cont.	
Initial Licensing	Y	N	N	• FHA in Fuel Building • 24 hours Decay • Regulatory Guide 1.25 Assumptions
35	N	Y	N	• New Analysis – FHA in Containment • 70 Hour Decay Time • 12 Vent and Drain Lines ($\leq$70.2 cfm leakage) • Leakage Rate = L_a + 70.2 cfm
85	N	N	Y	• New Analysis – FHA in Containment • 11 day Decay Time • No Containment Integrity • Regulatory Guide 1.25 Release Assumptions
110	Y	Y	Y	• Revised Analyses – USQ • RPF increased from 1.5 to 1.65 • I-131 Gap Fraction increased from 0.1 to 0.12 • Revised Release Rate for 11 day case • Assumed 3100 MWt (Power Uprate)
113	N	N	Y	Removed requirements for fuel building >11 days. Actual analysis not affected since the containment case for 11 days bounds an FHA in the fuel building >11 days.
114	Y	Y	Y	RBS Power Uprate. Analyses were not revised since Amendment 110 analyses conservatively accounted for power uprated conditions.
119	N	N	Y	Based on TSTF-51. Allowed containment equipment hatch to be opened >11 days. Analysis not revised since Amendment 110 was bounding.

Question:

8. Provide the amounts of fission product activities (in curies) released to the environment from the plant condenser following the postulated control rod drop accident.

Response:

The CRDA AST analyses were summarized in Attachment 9 of the RBS submittal. RBS evaluated two cases:

1. A design basis case based on Regulatory Guide 1.183 assumptions. Source term is conservatively based on 100% power operations with activity released via the plant condenser. The release rate assumed was 1 volume % per day in accordance with Regulatory Guide 1.183, Appendix C.

2. A lower power event. While the source term is based on full reactor power, the release in this event is based on the assumption that the plant Mechanical Vacuum Pumps (MVP) is operating (MVP can not be operated at >~5% reactor thermal power, or 145 MWt). This limited CRDA does not result in a main steam line dose rate that would cause the main steam line radiation monitor to trip the MVP. The MVP are assumed to be manually isolated 20 minutes into the event.

The fission product activities released to the environment for both analyses are presented in Table 6 below.

Table 6
CRDA Activity Released to the Environment (Curies)

Isotope	100% Power Event	Low Power Event
Kr-85	4.09E+02	1.29E+02
Kr-85m	9.14E+03	2.88E+03
Kr-87	1.31E+03	4.92E+03
Kr-88	4.17E+03	7.40E+03
Rb-86	7.51E-04	2.41E-03
I-131	1.71E+02	1.12E+02
I-132	3.56E+01	1.54E+02
I-133	2.55E+02	2.33E+02
I-134	2.09E+01	2.20E+02
I-135	1.27E+02	2.16E+02
Xe-133	7.02E+04	2.36E+04
Xe-135	4.48E+03	3.00E+03
Cs-134	8.72E-02	2.74E-01
Cs-136	1.87E-02	6.04E-02
Cs-137	5.42E-02	1.70E-01

Question:

9. Discuss the control room habitability design and operational features providing justifications for the validity of the assumed unfiltered air in-leakage to the control room and providing reasonable assurance that the River Bend proposed changes will not have an adverse impact on the main control room habitability.

Response:

This was an item of discussion at the October 8, 2002 meeting between the USNRC and Entergy. In the meeting Entergy representatives stated that it was Entergy's intent to conduct a baseline Control Room boundary unfiltered in-leakage testing in accordance with NEI Guidance 99-03 at RBS. Entergy will perform the baseline in-leakage test by the end of the next operating cycle (RF12).

The NRC staff also asked Entergy to perform a sensitivity evaluation to determine the maximum amount of unfiltered in-leakage for which the resulting dose would still be within 10 CFR 50 Appendix A, General Design Criterion (GDC) 19 limits. The RBS current design basis in-leakage is only 10 cfm based on the estimated contribution from opening and closing of doors. The proposed accident analyses using alternate source terms assumed an unfiltered in-leakage value of 300 cfm. We have performed sensitivity calculations and determined that the maximum amount of unfiltered in-leakage that would still meet the GDC 19 limits is approximately 430 cfm. The 300 cfm value should provide ample margin while not causing overly conservative results with respect to the analytical MCR doses.

EOI is aware of several industry emergency ventilation system performance issues that have raised generic questions about a licensee's in-leakage assumptions for design basis analyses. Although the River Bend Station employs a rigorous design for maintaining Control Room habitability, we are aware that there may be elements of the design that may be potential sources of small amounts of unfiltered inleakage and we are continuing to evaluate the RBS design as recommendations are developed through the NEI task force efforts. However, we believe that any such inleakage would not cause the operator dose to exceed the limits of GDC 19. The RBS habitability zone and supporting HVAC systems have been evaluated against draft NEI guidance for determining susceptibility to large amounts of unfiltered in-leakage and found to be not susceptible. The results of our evaluation of the RBS design provides reasonable assurance that any unfiltered inleakage would be small (i.e., the resulting operator dose would be less than GDC 19 limits).

The main control room envelope boundary is designed with low leakage construction to minimize the potential for the infiltration of air into the main control room. The enclosed volume of the main control room envelope is 240,700 cu ft. The walls, floor, and roof are constructed of poured-in-place reinforced concrete which is essentially leak tight. The access doors are of airtight design with self-closing devices which shut the doors automatically following the passage of personnel. All cable and air duct penetrations are provided with a fire retardant seal which provides leak tight construction.

RBS has a positive pressure control room and has documentation that the control room pressure when in the emergency mode can be maintained at approximately 1.0" w. g. These actual results exceed the Technical Specification requirement of .125" w. g. and provide assurance of boundary leak tightness. Additionally, previous testing has demonstrated that adjacent areas are at a lower pressure than the required post accident

Control Room envelope pressure of 0.125" w. g. and therefore cannot infiltrate into the envelope. This testing was performed in response to Information Notices 86-76 and 88-61.

Maintenance guidelines and Engineering and Operations review of work packages maintain the habitability zone. Work packages that breech the control room boundary are typically scheduled during periods when the boundary is not required (outages). Routine surveillance testing demonstrates continued leak tightness of the envelope boundary by performing a positive pressure test every 18 months. In addition, River Bend has installed permanent plant instrumentation that monitors Control Room pressure and alarms in the Control Room should that pressure decrease to a preset value. The Control Room envelope leak tightness is verified every 18 months by performance of a surveillance pressure test as required by Technical Specification SR 6.7.4.2.

Under normal plant conditions, outside air enters the main control room through the local outside air intake located on the roof of the control building. During accident conditions, fresh air may be drawn in through the remote air intake. Measurements taken from the radiation monitor in the air supply duct allow operators to select the least radioactive air intake. During a LOCA or upon high radiation detection, the outside air supply is automatically diverted through the main control room charcoal filter as a precaution regardless of outside air quality. During the emergency mode of operation, the CRFA System is designed to slightly pressurize the control room to at least one-eighth inch water gauge positive pressure with respect to outside atmosphere to prevent unfiltered inleakage. The configuration at RBS is such that 2000 CFM of recirculated air is mixed with 2000 CFM of outside air prior to being processed by the filter train and returned to the control room volume. This recirculation of the control room air through filters aids in reducing the amount of contaminants in the space and therefore reduces operator exposure.

All ductwork and HVAC equipment that is used to support the Control Room environment post accident is located inside the Control Room habitability envelope and any inleakage would be filtered or "clean air". The envelope includes the Control Room air conditioning units, the charcoal filter trains, and the suction and recirculation ductwork. A portion of ductwork for the local (normal) and remote air intakes is located outside the envelope. During the emergency mode of operation, the normal intake is isolated by two isolation valves. These isolation valves are identical to some containment isolation valves. Previous testing has demonstrated that any leakage across the normal outside air intake valves would leak into the suction of the charcoal filter trains and thereby be treated by the charcoal and HEPA filters.

The RBS design review did identify one potential source of small amounts of unfiltered inleakage. The Control Room smoke removal fan is part of the CR HVAC system and has ductwork that penetrates the habitability envelope and connects to the suction of the CR air conditioning units. This pathway is normally isolated by two redundant, low leakage (ANSI N509 Class III) isolation dampers. Athough these dampers are normally closed, they will automatically close on an isolation signal, if opened. These dampers would be subject to a pressure of 1.4 inches w.g. during a DBA situation. These dampers are rated per ANSI N509 to limit leakage to 148 cfm at a fan rated pressure of 13.4 inches w.g.. In addition, the vendor supplied rating for these dampers is 100 cfm at 13.4 inches w.g. The calculated leakage through a single damper at the DBA expected pressure of 1.4 inches w.g. is only 32 cfm. The closure of both of the dampers (which are normally closed) would reduce in-leakage even further. Although the actual leakage rate has not been tested, there is reasonable assurance that the dampers would limit leakage to within GDC 19 limits. RBS will test and quantify the actual leakage across these two dampers as part of resolving the habitability issues. We expect to complete this component testing by January 31, 2003.

Attachment 2

To

RBG-46053

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
RBS will test and quantify the actual leakage across the two dampers in the smoke removal system as part of resolving the habitability issues. We expect to complete this component testing by January 31, 2003.	X		01/31/2003
In the meeting Entergy representatives stated that it was Entergy's intent to conduct a baseline Control Room boundary unfiltered in-leakage testing in accordance with NEI Guidance 99-03 at RBS. Entergy will perform the baseline in-leakage test by the end of the next operating cycle (RF12).	X		RF12 (Fall 2004)