ENTERGY

Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61
P.O. Box 220
St. Francisville, LA 70775
Tel 504 336 6225
Fax 504 635 5068

James J. Fisicaro
Director
Nuclear Safety

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Document Control Desk
U. S. Nuclear Regulatory Commission
Mail Stop P1-37
Washington, D. C. 20555

Subject: River Bend Station - Unit 1
Docket No. 50-458
Licensing No. NPF-47
Comments on Preliminary Accident Sequence Precursor Analysis of Event at RBS

File Nos. G9.5, G9.42

References: RBC-46043, "Review of Preliminary Accident Sequence Precursor Analysis of Event at River Bend Station," dated May 8, 1995

RBF1-95-0140
RBG-41598

In the referenced letter, the NRC requested comments on the 1994 Precursor Report. Our comments on the report are included in Attachment 1 to this letter.

Also included as Attachment 2 is the River Bend Station (RBS) Safety Analysis of Scram #94-01 which occurred on September 8, 1994. The RBS analysis was performed using the appropriate RBS-specific Probabilistic Safety Assessment (PSA) models. This analysis, provided for information only, indicates that the RBS analysis is in general agreement with the NRC's Accident Sequence Precursor (ASP) analysis.

If you have any further questions, please contact Mr. Guy Davant of my staff at (504) 336-6223.

Sincerely,

James J. Fisicaro

JJF/ghd
attachments

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cc: U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

NRC Sr. Resident Inspector
P. O. Box 1051
St. Francisville, LA 70775

Mr. David L. Wigginton
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
M/S OWFN 13-H-3
Rockville, MD 20859

RBS Comments on the Preliminary Accident Sequence Precursor Analysis

As requested by the NRC in RBC-46043, Entergy Operations, Inc. (EOI) provides the following comments on the preliminary Accident Sequence Precursor (ASP) analysis for RBS. The appropriate ASP section/table is denoted followed by the associated comment(s).

Section A.1.4, Modeling Assumptions

1. Operations personnel could have recovered the feedwater system (FWS), if necessary to mitigate the event. This system failed due to the slow transfer of plant electrical loads to off-site power sources. All FWS pumps and valves were operable.

2. [illegible] trains of the control rod drive system (CRD) must be manually started for adequate initial reactor cooling. CRD is recoverable in this event. The operators could have either manually opened the CRD flow control valves or changed the flow control valve control circuit fuses (which blew because of the slow transfer) in time to use CRD as an injection source in this event. Therefore, CRD should be modeled as available, with appropriate recovery factors.

Table A.1.1, Definitions and Probabilities for Selected Basic Events for LER 458/94-023

1. Do the unavailability numbers for High Pressure Core Spray (HPCS), Residual Heat Removal (RHR), Automatic Depressurization System (ADS), etc. include terms for maintenance unavailability? If so, the analysis should reflect that these systems were available and not out of service due to maintenance activities.

Table A.1.2, Sequence Conditional Probabilities for LER 458/94-023

2. For Transient Sequence 7 denoted in the referenced NRC letter, please note that RBS does not have an RHR containment spray subsystem. RBS has containment unit coolers which are independent of RHR, but dependent on the Normal/Standby Service Water system. This is a plant-specific difference between RBS and the generic BWR/6 model.

Probabilistic Safety Analysis
of
Scram #94-01
(September 8, 1994)

RBS, as part of its evaluation of Scram #94-01, performed an analysis of core damage probability associated with this event. This analysis was performed using the RBS plant-specific PSA. Assumptions included:

- A transient initiator with loss of normal service water, loss of feedwater/condensate, loss of instrument and closure of the main steam isolation valves (MSIVs).
- Reactor Core Isolation Cooling system (RCIC) failed due to overspeed.
- No loss of off-site power, no loss of Reactor Primary Containment Cooling Water system (CCP), etc.
- Emergency Core Cooling Systems (ECCS) were not removed from service due to maintenance activities.
- Recovery from slow transfer is approximately equal to recovery of the Power Conversion System (PCS) modeled in NUREG/CR-4550, page 8-46.
- Standby Service Water (SSW) train "A" flow was sufficient to supply the necessary plant loads since adequate flow was available and operators were able to quickly open SSW pump "A" discharge valve 1SWP*MOV40A. This assumption is supported by the use of RHR "A" for suppression pool cooling.

RBS re-quantified the appropriate transient sequences and added appropriate recovery factors. Based on the quantification, the probability of core damage given the above scram is 1.21E-5 compared to the 6.0E-5 value presented in the NRC letter. The core damage frequency due to a "normal" scram (all systems necessary to mitigate accident consequences are available with normal maintenance availability assumptions) is 5.4E-8/yr per the Individual Plant Evaluation (IPE). Normal scram frequency is 2 scrams/year. Therefore, the probability of core damage during a normal scram is 2.7E-8. The model used in the referenced NRC letter should indicate the same relative change in core damage probability.