Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

September 15, 2006

TVA-BFN-TS-431
TVA-BFN-TS-418

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D. C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
	)	50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418 - EXTENDED POWER UPRATE (EPU) - RESPONSE TO ROUND 9 REQUESTS FOR ADDITIONAL INFORMATION (TAC NOS. MC3812, MC3743, AND MC3744)

By letters dated June 28, 2004 (ADAMS Accession No. ML041840109) and June 25, 2004 (ML041840301), TVA submitted applications to the NRC for EPU operation of BFN Unit 1, and BFN Units 2 and 3, respectively. On September 1, 2006, the NRC staff issued the Round 9 requests for additional information (RAIs) on the EPU amendment requests. This submittal responds to NRC staff questions APLA-25 and 26 (Unit 1), APLA-27 and 28 (Units 2 and 3), and ACVB-62 through 67 (Units 1, 2 and 3).

Enclosure 1 to this letter provides TVA's responses to the RAI questions identified above. Enclosure 2 to this letter is an evaluation referenced in the response to RAI APLA-26 (Unit 1) / APLA-28 (Units 2 and 3).

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TVA has determined that the additional information provided by this letter does not affect the no significant hazards considerations associated with the proposed TS changes. The proposed TS changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

No new regulatory commitments are made in this submittal.

If you have any questions regarding this letter, please contact me at (256)729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 15th day of September, 2006.

Sincerely,

William D. Crouch

William D. Crouch
Manager of Licensing
and Industry Affairs

Enclosures:
1. Responses to Round 9 RAI Questions
2. Evaluation of Frequency of General Transient Scenarios Potentially Requiring Containment Overpressure (COP), Revision 1

Enclosures

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418 -
EXTENDED POWER UPRATE (EPU) -
RESPONSE TO ROUND 9 REQUESTS FOR ADDITIONAL INFORMATION
(TAC NOS. MC3812, MC3743, AND MC3744)

RESPONSES TO ROUND 9 RAI QUESTIONS

This enclosure provides a partial response to the NRC staff's September 1, 2006, Round 9 Requests for Additional Information (RAIs) regarding the applications for extended power uprate of BFN Units 1, 2, and 3. Specifically, responses are provided to NRC questions APLA-25 and 26 (Unit 1), APLA-27 and 28 (Units 2 and 3), and ACVB-62 through 67 for Units 1, 2 and 3.

NRC Request APLA-25 (Unit 1) / 27 (Units 2 and 3)

In various correspondence the NRC staff has noted different values for the suppression pool (SP) bulk temperature limit:

a. Section 4.8.6.2, Page 4.8-4 of the updated final safety analyses report (UFSAR) states a limit of 177 degrees Fahrenheit (°F), based on an analysis of the torus attached piping.

b. The limit of 177°F was used in the previous 5 percent power uprate for Units 2 and 3 (ADAMS Accession No. ML042670045).

c. The draft Unit 1 Fire Protection Program Report (ADAMS Accession No. ML060620424) provides various limits as follows:

   i. Page 301- The design limit is 281°F.

   ii. Page 309 - The residual heat removal (RHR) pump seals were rated for 160°F, but have been re-evaluated for 215°F.

d. Table 4-1 of Enclosure 4 of the submittals dated June 28 and 25, 2004, uses the 281°F limit. Provide the correct SP bulk temperature limit for evaluating the proposed containment accident pressure (CAP) credit.

TVA Response to APLA-25 (Unit 1) / 27 (Units 2 and 3)

A discussion of the listed parameter values is provided below:

i. The design temperature of the drywell and suppression pool (pressure suppression chamber) structures is 281°F. This value is not changed by EPU and was not changed by the 105% power uprate on Units 2 and 3. The design temperature of 281°F is reflected in UFSAR Table 5.2-1, EPU PUSAR Table 4-1 (Limit), NEDC-32751P Table 4-1 (October 1, 1997, 105% original licensed thermal power (OLTP) submittal), and the Fire Protection Report.

ii. The current peak suppression pool temperature that was determined by the DBA-LOCA containment analysis for 105% OLTP is 177°F. This value of 177°F for 105% OLTP is reflected in the UFSAR, EPU PUSAR Table 4-1 (Units 2 and 3 current rated thermal power) and NEDC-32751P Table 4-1 (uprated power). This analyzed value changes for EPU. The EPU peak suppression pool temperature that was determined by the DBA-LOCA containment analysis for EPU conditions is

187.3°F. The value of 187.3°F for EPU is reflected in EPU PUSAR Table 4-1.

The torus attached piping is evaluated for pipe stresses including the peak suppression pool temperature for a DBA-LOCA. As currently reported in the UFSAR for 105% power, this value is 177°F. For EPU, this value increases to 187.3°F. As described in PUSAR Section 3.11, the torus attached piping has been evaluated for EPU conditions. This evaluation includes the increase in peak suppression pool temperature for the DBA-LOCA containment analysis. As part of the implementation of EPU following NRC approval, the UFSAR will be revised to reflect the EPU value of 187.3°F.

The current Units 2 and 3 (105% Power) NPSH evaluations for the long-term DBA-LOCA have been performed based on the 177°F peak suppression pool temperature for the DBA-LOCA containment analysis for 105% Power. The EPU NPSH evaluations for the long-term DBA-LOCA have been performed based on the 187.3°F peak suppression pool temperature for the DBA-LOCA containment analysis for EPU.

iii. The RHR pump seals are rated to 215°F for continuous operation, and from 215°F to 400°F for up to two days of operation. These values are not changed by EPU and were not changed by the 105% power uprate on Units 2 and 3. The rated value of 215°F is reflected in the Fire Protection Report Section 4.2.5. The discussion provided in the Fire Protection Report reflects the current (105% Power) peak suppression pool temperature for the Appendix R analysis of 212°F. As part of the implementation of EPU following NRC approval, the Fire Protection Report (including the discussion of the RHR pump seals) will be revised to reflect the EPU value of 227°F for the EPU Appendix R analysis. Similar to the current discussion in the Fire Protection Report, the suppression pool water temperature during the 10 minutes that the emergency equipment cooling water (EECW) is not available does not exceed the 215°F rating of the seals.

NRC Request APLA-26 (Unit 1) / 28 (Units 2 and 3)

Analysis (e.g., the August 4, 2006 submittal) indicates that containment accident pressure (CAP) credit is required to ensure adequate net positive suction head (NPSH) to the RHR pumps during an Appendix R scenario. The NRC staff understands that CAP credit is required for the pre-EPU plant as well as for the post-EPU plant. The Fire Protection Program Report defines the Appendix R scenario as a fire that results in a total loss of high-pressure makeup sources (feedwater (FW), high pressure coolant injection, and reactor core isolation cooling, followed by manual depressurization using three S/RVs and operation of one RHR pump and its associated heat exchanger in low pressure coolant injection (LPCI) mode (i.e., no suppression pool cooling, (SPC)).

For transient initiating events (e.g., loss of FW), the probabilistic risk assessment (PRA) credits manual depressurization using the S/RVs and use of either core spray (CS) or LPCI, along with SPC, upon the failure of all high-pressure makeup sources. The PRA also includes sequences initiated by transient events that lead to multiple stuck-open S/RVs (e.g., loss of FW and subsequent MSIV closure, which causes the S/RVs to open, followed by subsequent failure of the S/RVs to reclose). The previous risk evaluation of the proposed CAP credit does not address these types of accident sequences.

Provide a risk evaluation of the proposed CAP credit that includes the increase in core-damage frequency and, large early release frequency due to sequences that are initiated by transient events that lead to either (a) manual depressurization via the S/RVs and use of CS or LPCI upon the total loss of high-pressure makeup sources, and (b) sequences that are initiated by transient events that lead to multiple stuck-open S/RVs.

TVA Response to APLA-26 (Unit 1) / 28 (Units 2 and 3)

BFN Credited Containment Overpressure

The BFN design and licensing basis events have been evaluated to determine which events would require credit for available containment pressure to ensure adequate NPSH for the low pressure ECCS pumps (RHR and CS). These events are evaluated using deterministic analyses with inputs, assumptions, and methodologies that are prescribed by the BFN design and licensing basis for these specific events. Based on the deterministic analyses, TVA has concluded that credit for available containment pressure to ensure adequate NPSH for the low pressure ECCS pumps is required for the DBA-LOCA and three special events (i.e., Appendix R, ATWS, and SBO).

The risk associated with these events (i.e., DBA-LOCA, Appendix R, ATWS, and SBO) was addressed in TVA letter dated July 21, 2006 (ADAMS Accession No. ML062090071). Enclosure 2 of that letter provided the BFN EPU Containment Overpressure (COP) Credit Risk Assessment which addressed the DBA-LOCA, ATWS, and

SBO events. The response to RAI question APLA-24/26 in that letter provided a qualitative evaluation of the risk associated with the Appendix R event.

<u>Additional PRA Sequences</u>

Event sequences outside of the licensing basis can be postulated which would result in elevated suppression pool temperatures and challenge RHR and CS pump NPSH. Such sequences would typically involve multiple failures of decay heat removal functions. Given these event sequences, any containment pressure present will act to offset the effect of increasing pool temperature, improve available NPSH, and influence the outcome of the event. However, consideration of containment pressure in such sequences outside of the design and licensing bases does not constitute credit for COP.

To address the NRC request, TVA has performed an evaluation of events (other than LOCA and ATWS) leading to either depressurization using SRVs upon loss of high pressure makeup or multiple, stuck-open SRVs to quantify the impact on PSA results if ECCS pump NPSH and COP are considered. An upper bound frequency of those scenarios resulting from general transient scenarios (i.e., other than large LOCA and ATWS) was determined. This evaluation is provided in Enclosure 2 of this submittal. The results of the evaluation determined that based on the upper bound delta frequency, this consideration is non-risk significant.

<u>NRC Request ACVB-62</u>

The August 4, 2006 response to Request for Additional Information (RAI) Risk Assessment Containment & Ventilation Branch (ACVB) 37/35 states that, for the CS pump, the operator is instructed to maintain flow less than 4000 gallons per minute (gpm) and within the NPSH limit curves. However, for determining adequate NPSH, it is assumed that the operator would reduce flow in response to the NPSH limit curves, but not less than 3125 gpm.

It appears that at a flow rate of 4000 gpm and the peak calculated suppression pool temperature, the pumps are in the acceptable region of the Emergency Operating Instruction NPSH limit curves. Therefore, explain what prompts the operator to reduce flow to 3125 gpm. If the operator can operate acceptably at 4000 gpm, address why shouldn't this more conservative flow rate be used in the NPSH analyses.

TVA Response to ACVB-62

In the long-term LOCA event, the operator will control ECCS pump flow in accordance with the EOIs and within the NPSH limit curves based on plant symptoms. The NPSH analysis for long-term LOCA shows that at the peak suppression pool temperature, margin is available between the required and available wetwell pressures. The margin indicates that adequate NPSH would be available at 4000 GPM and the operator would not and should not be prompted by the EOIs to reduce core spray flow.

In contrast to the symptomatic EOIs, the safety analysis that forms the basis for COP credit is based on worst case conditions. The objective is to demonstrate that the ECCS pumps will be able to perform their safety function given bounding events and worst case assumptions. Consistent with this objective, the pump flow rates used are the minimum acceptable flows needed for the safety function plus a conservative margin. This flow is 3125 GPM in the case of core spray. By performing the analysis at this value, the safety function is assured and COP margin is established. At flows above this value, operator action will ensure adequate NPSH is maintained as dictated by the symptoms as they exist.

NRC Request ACVB-63

In the July 21, 2006 response to RAI APLA 24/26, five fire areas are described. For those fire areas for which the safety analysis depends on RHR pumps (control room and turbine building), 2 RHR pumps are said to be available. Address why only one RHR pump is credited for the Appendix R analyses and NPSH analyses.

TVA Response to ACVB-63

In the July 21, 2006 response to RAI Probabilistic Risk Assessment Licensing Branch A (APLA) 24/26 (ML062090071), a qualitative evaluation of realistic postulated fires was provided to assess the likelihood of requiring containment overpressure to maintain adequate NPSH for the low pressure ECCS pumps. For each of the fire areas described, the assessment considered the equipment that would be expected to be available to offset the need for containment overpressure. As discussed in that response, two RHR pumps would be available for certain fire areas.

In contrast to the realistic approach discussed in the July 21, 2006, response, the Appendix R analysis is a bounding analysis based on worst case conditions. This evaluation is based on prescriptive deterministic evaluations using conservative inputs and assumptions that results in a minimum complement of available safe shutdown system (SSDS) equipment. The minimum SSDS includes the equipment necessary for the alternate injection mode with one RHR pump.

NRC Request ACVB-64

Enclosure 4 of the August 4, 2006 letter contains Calculation MDQ099920060011, Transient NPSH/ Containment Pressure Evaluation of RHR and CS Pumps. For the short term loss of coolant accident (LOCA) response, Figure 7.5 of Calculation MDQ099920060011 shows that the wetwell pressure required is less than the wetwell pressure available for the RHR pumps pumping into the broken recirculation loop. TVA indicated this was acceptable based on RHR pump tests reported in Enclosure 2 to a May 21, 1976 TVA letter to the NRC. A margin of 9 feet was shown to be available in these tests relative to the required NPSH based on a 3 percent head drop.

(i) Provide the margin between the lowest NPSH value of the cavitation tests reported in the May 21, 1976 letter and the reduced required NPSH values used in Tennessee Valley Authority (TVA) Calculation MDQ099920060011.

(ii) Discuss the difference between the required NPSH and the available NPSH at 600 seconds.

(iii) Describe how the required NPSH value of 28.4 ft in Figure 7.5 of Calculation MDQ099920060011 was obtained.

TVA Response to ACVB-64

(i) The lowest measured NPSH value at 12,000 gpm shown in Table 3 of the referenced 1976 report is 25.0 ft. This is 3.4 ft. less than the requirement applied in the current analysis (28.4 ft. minus 25.0 ft.). Applying this value in the current analysis (instead of 28.4 ft.) would yield a positive NPSH available margin of 1.4 ft. (26.4 ft. minus 25.0 ft.) at 600 seconds.

(ii) At 600 seconds, the required NPSH shown in Figure 7.5 is 28.4 ft., and the available NPSH is 26.4 ft. (reference Table 6.2-1 of Calculation MDQ099920060011, Rev. 1, in TVA submittal of August 31, 2006).

(iii) The short-term LOCA NPSH requirements were obtained from the vendor data at 0.16667 hours (600 seconds). First, a requirement curve at 10,500 gpm was developed by interpolation between the vendor curves for 10,000 and

11,000 gpm. Then it was assumed that the required NPSH at 11,500 would increase proportionately to the increase from 10,000 to 10,500 gpm. (reference Attachment 1, page 1, of Calculation MDQ099920060011, Rev. 1, in TVA submittal of August 31, 2006).

NRC Request ACVB-65

Table 10-2 of Enclosure 4 to the June (28)25, 2004, submittal, NEDC-33101P, DRF 0000-0010-9439, Browns Ferry Unit 1 Safety Analysis Report for Extended Power Uprate (PUSAR) [NEDC-33047P, DRF 0000-0011-1328, Revision 2, Browns Ferry Units 2 and 3 Safety Analysis Report for Extended Power Uprate (PUSAR)], shows that the peak drywell air temperature due to a steam line break (336°F) exceeds the containment shell design temperature limit (281°F). Verify that the shell temperature itself remains below the 281°F design limit.

TVA Response to ACVB-65

The highest peak airspace temperature for the postulated steam line break is 335.4°F for EPU conditions. The resultant peak shell temperature is 277.1°F which is below the design value of 281°F.

NRC Request ACVB-66

Provide the maximum RHR and core spray pump seal temperatures. If less than the calculated peak suppression pool temperatures, address why this is acceptable.

TVA Response to ACVB-66

The RHR pump seals are discussed in the above response to RAI question APLA-25/27. The rating of the RHR pump seals is greater than the expected suppression pool temperatures for the events which include RHR pump operation (DBA-LOCA, Appendix R, ATWS, and SBO).

The CS pump seals are rated for 210°F. This is greater than the expected suppression pool temperatures for the event which includes CS pump operation (DBA-LOCA).

NRC Request ACVB-67

Provide the maximum acceptable temperature of the piping attached to the torus. If less than the maximum suppression pool water temperature, address why is this acceptable.

TVA Response to ACVB-67

Evaluation of the torus attached piping is addressed in PUSAR Section 3.11. Pipe stress evaluations for the affected piping are performed based on temperature and pressures associated with operating modes for each piping segment. The calculated pipe stresses are then compared to applicable code allowables. For EPU, these evaluations were performed utilizing the increased peak suppression pool temperatures associated with EPU conditions. All piping is below the code allowables of the present code of record.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418 -
EXTENDED POWER UPRATE (EPU) -
RESPONSE TO ROUND 9 REQUEST FOR ADDITIONAL INFORMATION
(TAC NOS. MC3812, MC3743, AND MC3744)

EVALUATION OF FREQUENCY OF GENERAL TRANSIENT SCENARIOS POTENTIALLY REQUIRING CONTAINMENT OVERPRESSURE (COP), REVISION 1

This enclosure provides TVA's evaluation of certain event sequences where credit for containment overpressure may be required to ensure adequate net positive suction head for low pressure ECCS pumps.

Evaluation of Frequency of General Transient Scenarios
Potentially Requiring Containment Overpressure (COP), Revision 1

Prepared by ~~D. A. Bolig~~ FOR Mitch Waller
VIA TELECON

Reviewed by R. Abbas Rashid Abbas

Approved by J.D. Wolcott 9-13-06 J.D. Wolcott

NRC Request APLA-26 (Unit 1) / 28 (Units 2 and 3) from the NRC RAI Round 9:

Analysis (e.g., the August 4, 2006 submittal) indicates that containment accident pressure (CAP) credit is required to ensure adequate net positive suction head (NPSH) to the RHR pumps during an Appendix R scenario. The NRC staff understands that CAP credit is required for the pre-EPU plant as well as for the post-EPU plant. The Fire Protection Program Report defines the Appendix R scenario as a fire that results in a total loss of high-pressure makeup sources (feedwater (FW), high pressure coolant injection, and reactor core isolation cooling, followed by manual depressurization using three S/RVs and operation of one RHR pump and its associated heat exchanger in low pressure coolant injection (LPCI) mode (i.e., no suppression pool cooling, (SPC)).

For transient initiating events (e.g., loss of FW), the probabilistic risk assessment (PRA) credits manual depressurization using the S/RVs and use of either core spray (CS) or LPCI, along with SPC, upon the failure of all high-pressure makeup sources. The PRA also includes sequences initiated by transient events that lead to multiple stuck-open S/RVs (e.g., loss of FW and subsequent MSIV closure, which causes the S/RVs to open, followed by subsequent failure of the S/RVs to reclose). The previous risk evaluation of the proposed CAP credit does not address these types of accident sequences.

Provide a risk evaluation of the proposed CAP credit that includes the increase in core-damage frequency and, large early release frequency due to sequences that are initiated by transient events that lead to either (a) manual depressurization via the S/RVs and use of CS or LPCI upon the total loss of high-pressure makeup sources, and (b) sequences that are initiated by transient events that lead to multiple stuck-open S/RVs.

TVA Response:

Introduction

This evaluation was performed in response to a NRC Request for Additional Information (RAI) associated with utilizing containment overpressure (COP) to satisfy the net positive suction head (NPSH) requirements for the low pressure emergency core coolant system (ECCS) pumps during specified sequences.

Summary of Results

The risk assessment evaluation is based upon the current BFN Unit 1 Probabilistic Risk Assessment (PRA) internal events model, which includes internal flooding initiating events. To address NRC Request APLA-26/28, this evaluation was performed by assuming that all scenarios with less than 2 trains of RHR cooling for the suppression pool following loss of main condenser heat sink and either requiring RPV depressurization or resulting in 2 or more stuck open relief valves result in core damage. An upper bound delta frequency of these scenarios (other than large LOCA and ATWS) is estimated as 1.51E-7 per reactor year. Assuming that all of this frequency is assigned as core damage results in a non-risk significant condition (i.e., less than 1E-6 increase in CDF). Large early release delta frequency is then estimated to be no more than 2.56E-8, again resulting in a non-risk significant condition (i.e., less than 1E-7 increase in LERF).

PSA Evaluation

The risk associated with the events for which credit for containment overpressure is required (DBA-LOCA, Appendix R, ATWS, and SBO) has been previously evaluated. This study evaluates additional transient events which involve heatup of the suppression pool via SRVs. Bounding CDF and LERF values were determined with the following simplifying assumptions:

1. Suppression pool heating is not considered if turbine bypass is available.

2. Suppression pool temperature will not be high enough to require COP provided that at least 2 RHR pumps/heat exchangers are available.

3. If COP is required, core damage is assumed.

All initiating events were included except LOCA and ATWS events.

As a bounding level of significance, the PSA plant model success criteria were changed to require two trains of suppression pool cooling for success. This was incorporated into a cloned copy of the U10806 model by adding a macro named RHRCASE1 in event tree module LPGT3 with the following logic:

```
RHRCASE1  HXASUP*(HXBSUP+HXCSUP+HXDSUP+U2X=S)
          +HXBSUP*(HXCSUP+HXDSUP+U2X=S)
          +HXCSUP*(HXDSUP+U2X=S)
          +HXDSUP*U2X=S
```

This requires that two trains of RHR heat removal are available for the indicated conditions. This macro was then inserted as an "AND" condition into the NCDSORV and NCDLVPRES macros to define success conditions when 2 or more relief valves are stuck open (SORV2) and when LPCI or core spray are required for makeup (i.e. RPV depressurization has taken place and suppression pool cooling is required).

Due to the number of scenarios retained in the database when RHRCASE1 was "ANDED" with the NCDLVPRES macro in event tree module LPGT3, the following additional logic was provided in this event tree module:

...*(RHRCASE1+(HPI=S+RCI=S)*ORVD=B	COP not required when HPCI/RCIC available
+ RVC=SORV1	COP only for 2 or more SORV condition
+ INIT=FLRB3S)	Torus/suppression pool not credited (i.e., COP not applicable) for this initiating event

Also, since the containment response model assumes that these scenarios are not assigned to CDF, a "CLASS" for LERF evaluation is not assigned. Therefore, the split fraction logic for AL0 in event tree module CET1 is set to 1 (i.e., a default case is assigned).

The non-LLOCA and non-ATWS initiators were quantified with a corresponding CDF of 5.9962E-6 (increase from 1.4421E-6, or 4.5541E-6 total increase for the same initiators in the base Unit 1 model). The Unit 1 model U10806 was selected for this evaluation due to the ATWS initiators being separately evaluated, so that they could be readily separated from this evaluation.

One distinct family of scenarios emerged from this evaluation:

1. Loss of main condenser vacuum, followed by failure of RCIC and HPCI injection. This condition mandates an emergency RPV depressurization to allow low pressure injection (core spray/LPCI). This was then followed by independent failure of RHR pump trains A, C and D.

Rank	Initiator	Index	Frequency	Failed and Multi-State Split Fractions	Bin
1	LCV	69	1.0220E-006	//OG16F//DWF*/RVC0*MCDF*RCI1*HPI4*/FWSDF*RPA1*HXAF*RPC2*HXCF*RPD3*HXDF*/PCSF*/SAFEF*PCSRF*//ELF	NOLERF

As a point of reference, this scenario has a frequency of 1.02E-6 and invokes RHR pump failures with split fractions RPA1 (=2.33E-2), RPC2 (=0.347) and RPD3 (=0.225).

The failure of concern for this series of scenarios is the second RHR pump. For example, the failure rate for RPC2 is extremely high, even accounting for common cause failure, which would be expected to result in a second train failure rate of approximately 0.07.

This is partially explained by the fact that RHR failure rates are calculated with an interim top event, RPX, which evaluates failure of all four trains of RHR. The individual train failure rates are then calculated from this master fault tree. For example, RPD2 is calculated as RPX2S/RPX1, where RPX2S is the likelihood of two trains of RHR pumps in the same loop fail and RPX1 is the likelihood of a single train failing. The failure rate for two pumps would be RPX2S and the RPX1 term accounts for the use of the RPA1 split fraction earlier in the event tree scenario.

This is due to the assumed model simplification that the second RHR pump is in the same loop as the first one failed. Further evaluation revealed that the failure of two RHR pumps in a single train is dominated (94%) by both RHR pumps being unavailable due to loop testing (alignment RPIT1 – compare cutsets for interim split fractions RPX2S (= 8.20E-3, dominated by alignment RPIT1) and RPX2E (= 2.59E-4, two RHR pumps fail, RPIT1 does not contribute), more than a factor of 30 difference).

MODEL Name: U1COP2RH
Cause Table for Top Event RPX and Split Fraction RPX2S

PE Value of RPX2S = 8.2049E-03 Date : 15 JUN 2006 14:59

No...	Cutsets................................	Value.....	% Importance.	% Cumulative.	Alignment...
1	Always Failed	7.6591E-03	94.60467	94.6047	RPIT1
2	Always Failed	1.0792E-04	1.333020	95.9377	RPIV
3	MOVXC1FCV0740007 (FCV-74-7 TRANSFERS CLOSED)	9.6217E-05	1.188465	97.1262	NORMAL
4	HOVXC1HCV0740085 (HCV-74-85 TRANSFERS CLOSED)	4.4503E-05	0.549697	97.6759	NORMAL
5	HOVXC1HCV0670565 (VALVE 67-565 TRANSFERS CLOSED)	4.4503E-05	0.549697	98.2255	NORMAL

Discussion of Plant Response

In the plant, this scenario would be expected to develop with the control room operator responding to the plant trip condition, eventually placing the one available RHR train (RHR pump B in the scenario discussed above) in suppression pool cooling mode at a suppression pool temperature of 95 F (following plant trip with no turbine bypass – relief valves lift to discharge decay heat to the suppression pool). This condition would then be exacerbated by the failure of high pressure injection with HPCI and RCIC.

At this point, it should be apparent to the operator as to why two of the remaining pumps are unavailable (i.e., loop testing) and the shift supervisor would be expected to direct the plant operators to secure from loop testing and restore at least one additional RHR pump train to operability before the suppression pool exceeds 100° F (i.e., the point at which plant procedures direct the operator to put a second train of RHR on line in suppression pool cooling mode).

In other words, while this initial evaluation gives a bounding value, it is excessively conservative in that the recovery of at least one of the failed RHR trains (i.e., restoration from loop testing) prior to requiring it to be placed in service due to suppression pool heatup to the point of questioning COP credit to maintain required NPSH is quite likely.

Estimate of Conservatism in Calculated CDF

A more realistic evaluation of the level of significance of the scenarios resulting from other plant transients that could potentially require containment overpressure would then be:

4.5541E-6 / 30 = 1.51E-7 change in CDF

This evaluation remains bounding in that all scenarios with less than 2 trains of RHR cooling for the suppression pool following loss of main condenser heat sink and either requiring RPV depressurization or resulting in 2 or more stuck open relief valves are now assumed to result in core damage, whereas these would represent the maximum set of scenarios for which containment overpressure would potentially be an issue for consideration.

Also, it should be noted that, while the containment overpressure issue has been evaluated with the respective core spray and RHR pumps operating at near maximum flows, RPV makeup without a large break LOCA would be much less challenging, such that either makeup flow would be throttled to maintain level or injection would be cycled to prevent RPV overfill. This reduction in required flow would make pump NPSH requirements somewhat less restrictive.

Contribution to Large Early Release Frequency

Since the primary mode for core failure under the proposed conditions would be due to lack of RPV makeup following loss of NPSH to low pressure injection pumps, the primary LERF contributor would be to containment failure with the reactor at low pressure and no in-vessel recovery. This condition would be bounded by the base case distribution of LERF versus CDF (2.4319E-7/1.4421E-6 = 0.169). This would result in a LERF contribution of no more than

1.51E-7 x 0.169 = 2.56E-8 change in LERF

Again, this assumes that all potential COP scenarios result in core damage.