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A Review of the Oconee-3 Probabilistic Risk Assessment

Internal Events, Core Damage Frequency

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

A review of the Oconee-3 Probabilistic Risk Assessment (OPRA) was conducted with the broad objective of evaluating the contribution of the internally-generated accidents to the frequency of core damage. The review included a technical assessment of the assumptions and methods used in the OPRA study. The BNL staff reevaluated the main results of the study within the scope and general methodological framework, including both qualitative and quantitative analyses of accident initiators, and accident sequences which result in core damage. The effect of uncertainties was considered throughout the review process, and the uncertainty bands for the core damage frequency were quantified.

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This work was performed for the Reliability and Risk Assessment Branch (RRAB) of the U.S. Nuclear Regulatory Commission. Mr. E. Chelliah of RRAB was the technical monitor of the project. The authors wish to acknowledge F. Coffman, E. Chelliah, and A. Thadani for comments on the draft of this report.

Lastly, the authors would like to acknowledge the relentless effort and the high standard of quality of C. Conrad in editing and preparing this document for publication, and of S. Flippen for her excellent typing.

NOMENCLATURE

A	Large LOCA
ATWS	Anticpated transients without scram
B	Failure of RCS heat removal via the steam generators
B&W	Babcock & Wilcox
BNL	Brookhaven National Laboratory
BWST	Borated water storage tank
CAS	Compressed air system
CCS	Component cooling system
CCWS	Condenser circulating water system
CFT	Core flooding tanks
DHR	Decay heat removal
DPC	Duke Power Company
EPRI	Electric Power Reseach Institute
EFW	Emergency feedwater system
ESFAS	Engineered safeguard actuation system
FMEA	Failure modes and effects analysis
FSAR	Final safety analysis reports
HVAC	Heating, ventilating and air-conditioning systems
HPI	High pressure injection system
HPR	High pressure recirculation
HPSW	High-pressure service water system
ICS	Integrated control system
IREP	Interim Reliability Evaluation Program
K	Failure of the reactor protection system
L	Failure of recover RCS heat removal
LDST	Letdown storage tank
LMFW	Loss of main feedwater
LOCA	Loss of coolant accident
LOOP	Loss of offsite power
LPI	Low pressure injection system
LPR	Low pressure recirculation
LPSW	Low pressure service water system
MFW	Main feedwater system

NOMENCLATURE Continued

MPRA	Midland Probabilistic Risk Assessment
NRC	U.S. Nuclear Regulatory Commission
NSAC	The Nuclear Safety Analysis Center
OPRA	Oconee-3 Probabilistic Risk Assessment
PORV	Pilot-operated relief valve
PCS	Power conversion system
PWR	Pressurized water reactor
PPCS	Primary pressure control system
PRA	Probabilistic risk assessment
Q	Loss of RCS integrity
R	Steam generator tube rupture
RBCS	Reactor building cooling system
RBSS	Reactor building spray system
RCS	Reactor coolant system
RPV	Reactor pressure vessel
RPS	Reactor protection system
RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Application Program
RCWS	Recirculating cooling water system
S	Small-break LOCA
SSF	Standby Shutdown Facility
SGTR	Steam generator tube rupture
SRV	Safety relief valve(s)
T	Transient initiators
T_1	Reactor/turbine trip
T_2	Loss of main feedwater
T_3	Partial loss of main feedwater
T_4	Loss of condenser vacuum
T_{5SUBF}	Failure of offsite power due to 230-kV substation failure
T_{5FEEDF}	Failure of offsite power due to grid or feeder failure
T_6	Loss of air
T_7	Excessive feedwater
T_8	Spurious engineered safeguards actuation signal

NOMENCLATURE Continued

T_9	Steam-line break
T_{10}	Feedwater-line break
T_{11}	Loss of ICS power bus 3KI
T_{12}	Loss of low pressure service water
T_{13}	Spurious low pressurizer pressure signal
T_{14}	Loss of 4-kV switchgear 3TC
U_R	Failure of high pressure injection (SGTR)
U_S	Failure of high pressure injection (LOCA)
UST	Upper storage tank
U_T	Failure of core-heat removal by HPI cooling ('feed and bleed')
VS	Very small-break LOCA
W	Failure to establish RCS integrity
X_T	Failure to maintain long-term core-heat removal
Y_T	Failure to maintain RCS makeup supply

EXECUTIVE SUMMARY

This review of the Oconee-3 Probabilistic Risk Assessment (OPRA) by Brookhaven National Laboratory (BNL) was sponsored by the U.S. Nuclear Regulatory Commission (NRC). The Oconee-3 PRA was performed by the Nuclear Safety Analysis Center (NSAC) of the Electric Power Research Insititute, Duke Power Company (DPC), and other participating utility companies. The OPRA includes estimates of the frequency of accidents (internally and externally initiated events) that may lead to severe core damage, the frequency and characteristics of release of radionuclides, and the magnitude of the resulting public health effects. This review, which presents only an assessment of the frequency of "internally" generated plant "accidents" (including loss of offsite power) leading to core damage, was begun in January 1985 by members of the Risk Evaluation Group at BNL. A companion review of the frequency of "externally" initiated events was initiated at about the same time and the results are issued in Volume 2 of this report.

The broad objective of this review was to evaluate the OPRA with respect to the overall frequency of core damage, the main contributors to this frequency, and the associated parameter uncertainties, and to consider all these for core damage accidents initiated by functional events internal to the plant, as well as loss of offsite power. The review included a technical assessment of the assumptions and methods used in the OPRA study, as well as a reevaluation of the results of the study. This included both qualitative and quantitative analyses of accident initiators, data bases, human errors, and accident sequences which lead to core damage.

The review process included a site visit and a meeting with Duke Power Company and NRC. This review also benefited from the DPC responses to several BNL questions. The DPC staff was very helpful and cooperative throughout the course of the review.

The main conclusions of the review are the following:

a. Within the stated scope, the OPRA study is an excellent piece of work. The same tools were used as for the Reactor Safety Study (event trees/fault trees), but the OPRA also added to the state of the art.

b. The reviewers believe that, notwithstanding the differences discussed below, the OPRA study successfully identified the major failure combinations that can lead to core damage. The reviewers also believe that modification of the OPRA study to reflect the results of this review will give a more realistic portrayal of the major characteristics of the Oconee-3 plant.

c. The assessment of the core damage frequency resulting from "internally" initiated events performed in this review includes a modification made to the Oconee-3 plant after completion of the study of internal floods by the OPRA team, namely the addition of an automatic backup pressure service water (LPSW) cooling of the motors of the high pressure injection (HPI) pumps. The OPRA did not factor this modification into the internal events analysis. In addition, the BNL review also included the information that Oconee-3 has been operating with pressurizer PORV block valve closed for most of the time (about 80% of operating time). This was not taken into account in the OPRA.

d. The total frequency of core damage calculated in this review is equal to 9.3E-5/yr as compared to 5.4E-5/yr for the OPRA. The contributors to the core damage frequency by bin and by initiating event category are summarized in Tables 0.1 and 0.2; in both tables a comparison with the OPRA is also given. In Table 0.3, the dominant accident sequences in each bin are presented; for comparison, the OPRA dominant accident sequences are presented in Table 0.4. From these tables the following conclusions can be drawn:

d.1. The largest increase is present in Bin III (5.7E-5/yr in this review vs 3.0E-5/yr in the OPRA). This difference is primarily due to events caused by loss of instrument air (as an initiator or due to loss of offsite power). This difference in core damage (CD) frequency is mainly due to the assumption on time available for recovery of compressed air. In Oconee-3 a loss of instrument air causes the drainage of the upper storage tank (UST) to the condenser hotwell, because valve C-176, which is normally used as a means of hotwell makeup from the UST (hotwell level control system), fails open on loss of air. In the OPRA, between two and six hours were used for quantification of the probability of failure to recover air. In a meeting held at DPC to discuss BNL comments regarding the OPRA, it was verified that about one hour would be more appropriate for drainage of the UST into the hotwell, and therefore the time available for the operators to recover air or transfer the EFW suction from the UST (primary source) to the hotwell. This change in time modifies the probability of failure to recover air from the value used in the OPRA (5.5E-2) to the value used in this review (0.5); this latter value is based on the OPRA assessment for recovery of air together with the judgment of the reviewers. For more details on specific sequences, see Appendix A.

In this review the loss of instrument air (T_6) becomes the most dominant contributor to the core damage frequency, being responsible for 49% of the CD frequency due to transients with scram, and for 33% of the total. In the OPRA it is responsible for 11% of the CD frequency due to transients with scram, and 6% of the total.

Note that according to a letter from H. B. Tucker (DPC) to H. R. Denton (NRC) dated Sept. 20, 1985, the Duke Power Company has taken an interim measure, namely the closure of manual isolation block valve (C-175) upstream of the air-operated valve C-175, to prevent the potential drainage of the upper surge tank (UST) following a loss of instrument air. In the near future, according to the referenced letter, a modification to air-operated valve C-176 is planned. If this modification is taken into consideration, the BNL-calculated core damage frequency for Bin III will decrease from 5.7E-5/yr to about 3.1E-5/yr and the total CD frequency will be equal to 6.7E-5/yr instead of 9.3E-5/yr.

d.2. Table 0.2 also shows that the loss of low pressure service water transient (T_{12}) in the OPRA is the most important transient, being responsible for about 45% of the core damage frequency due to transients with scram (24% of the total CD). In this review it accounts for 29% of the core damage due to transients with scram (19% of the total CD).

It is explained in Appendix A that some degree of conservatism exists in this review, because of the assumption that if valve CCW-73 fails closed, a loss of LPSW to its most important loads, i.e., cooling of HPI and RCP pump motors and heat exchangers in the component cooling water system, will occur. However, a detailed pipe-flow calculation is needed to verify this assumption and this calculation is beyond the scope of this review.

Note also that if a recent DPC modification made in the discharge of the LPSW to the cooling of HPI pump motors (DPC Drawings Nos. PO-115B, Rev. 26, and PO-124D, Rev. 12) is considered, the contribution of loss of low pressure service water transient (T_{12}) to the total CD frequency will decrease from 1.8E-5/yr to about 4.0E-6/yr; i.e., the total CD frequency in this review will become 7.9E-5/yr instead of 9.3E-5/yr.

d.3. The differences in CD frequency for instrument air (T_6) and for loss of low pressure service water (T_{12}) accounts for 71% and 13% of the increase in the total CD frequency, respectively. The remainder of the difference in total CD frequency comes from small LOCAs (8%), ATWS (4%), and large LOCAs (3%).

e. The uncertainty in the frequency of core damage performed in this review is presented in Figure 0.1. The 90% probability range for the frequency of core damage in this review spans an interval slightly higher than one order of magnitude from 1.9E-5/yr (5% percentile) to 2.5E-4/yr (95% percentile). The uncertainty, as in the OPRA study, should be interpreted as being introduced by uncertainties in the values of the various parameters, given the modeling assumptions described in the main body of this report.

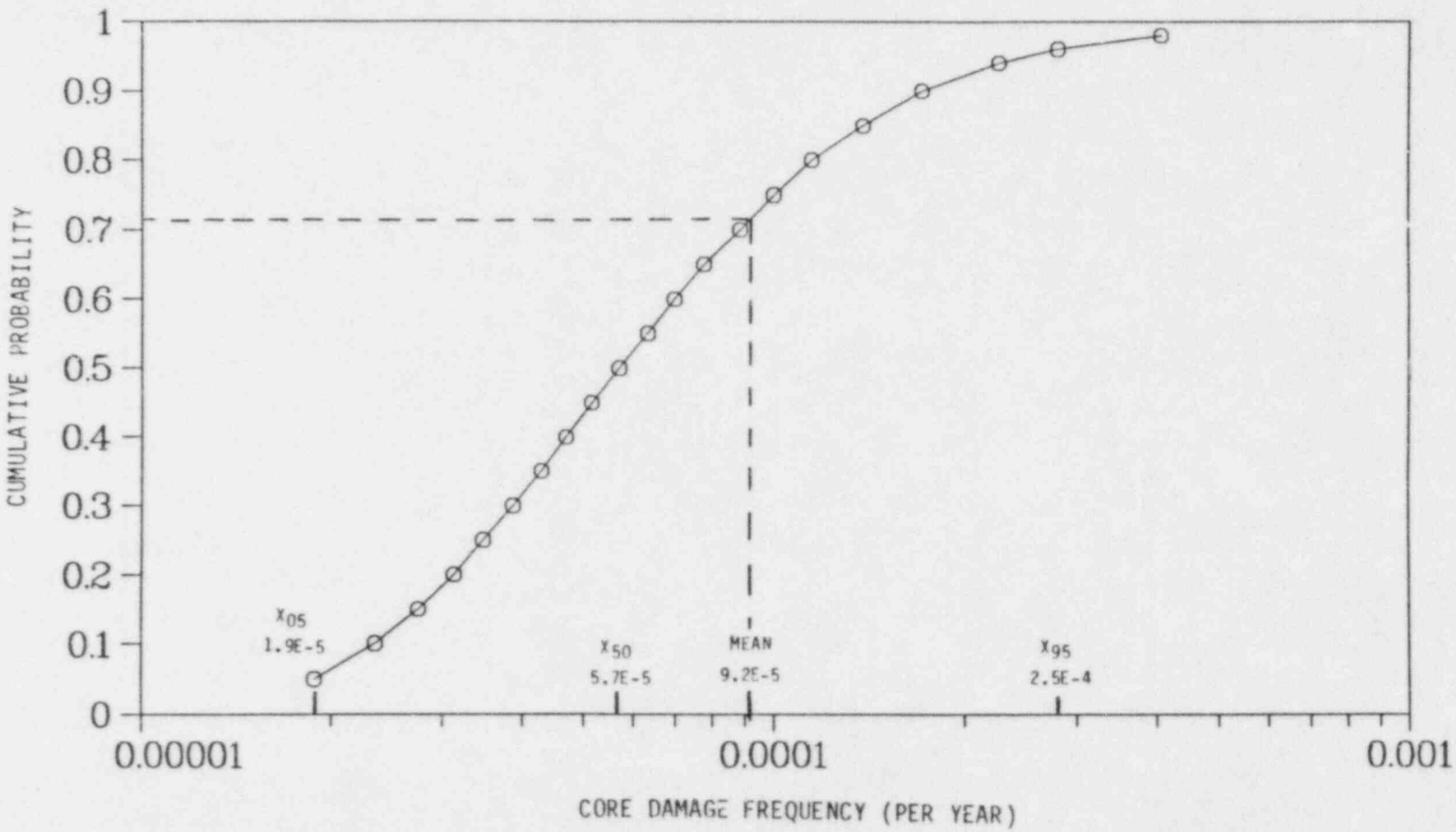

Figure 0.1 Cumulative probability for the frequency of core damage.

Table 0.1 Summary of Contributors to Core Damage Frequency for Internal Initiating Events

CM Bin	Initiating Event	Core Damage Frequency OPRA	Core Damage Frequency BNL
I	Pipe-break- and transient-induced small LOCA	6.5-6	8.4-6
	SGTR	1.3-6	1.2-6
	ATWS	1.8-8	1.2-8
	Total bin I	7.8-6	9.6-6
II	Pipe-break- and transient-induced small LOCA	1.1-6	6.7-6
	SGTR	1.4-6	2.1-6
	ATWS	1.8-8	1.2-8
	Total bin II	2.5-6	8.8-6
III	Transients	2.7-5	5.7-5
	ATWS	2.8-6	6.9-7
	Total bin III	3.0-5	5.7-5
IV	Transients	1.9-7	3.6-7
	Total bin IV	1.9-7	3.6-7
V	Large LOCA	1.4-6	1.5-6
	ATWS	1.7-6	3.6-6
	Total bin V	3.1-6	5.1-6
VI	Large LOCA	8.3-6	8.5-6
	ATWS	1.5-6	3.4-6
	Total bin VI	9.8-6	1.2-5
	Interfacing systems LOCA	1.4-7	1.4-7
	Total CD frequency	5.4-5	9.3-5

Table 0.2 Summary of Core Damage Frequency

Initiating-Event Category	Core Damage Frequency BNL	OPRA
Plant Transients		
Loss of Instrument Air (T_6)	3.1-5	3.2-6
Loss of Service Water (T_{12})	1.8-5	1.3-5
Feedwater Line Break (T_{10})	4.5-6	4.8-6
Loss of Offsite Power (T_5)	3.6-6	2.4-6
Loss of Reactor Trip (T)	1.7-6	1.2-6
Loss of Main Feedwater (T_2)	1.3-6	1.2-6
Other Transients	2.4-6	2.6-6
	6.2-5	2.9-5
Loss-of-Coolant Accidents		
Large Break (A)	1.0-5	9.0-6
Small Break (S)	9.2-6[a]	6.1-6
Reactor-Vessel Rupture	1.1-6	1.1-6
Total	2.0-5	1.6-5
Transients Without Scram (ATWS)	7.7-6	6.0-6
Steam Generator Tube Rupture (R)	3.3-6	2.7-6
Interfacing-System LOCA	1.4-7	1.4-7
Total	9.3-5	5.4-5

[a]Includes only LOCAs due to pipe breaks or spontaneous seal failures.

Table 0.3 BNL Review Summary of Core Damage Frequencies for Internal Events
- Total CD Frequency = 9.3E-5/yr.

Bin I Sequences			Bin II Sequences			Bin III Sequences			Bin IV Sequences			Bin V Sequences			Bin VI Sequences		
Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0E-6 (ABOUT 89% OF TOTAL FREQUENCY)																	
													ATWS	3.6-6			
[A]	S[illegible]$_S X_S$	5.4-6	[B]	T_6QX_S	3.2-6	[F]	T_6BU	2.9-5				[A]	VR	1.1-6	[B]	AX_A	4.8-6
			[C]	V_SX_S	1.9-6	[G]	$T_{12}BU$	1.8-5							[A]	AX_A	3.6-6
			[B]	RX_RO	1.5-6	[E]	$T_{10}BU$	4.8-6								ATWS	3.4-6
						[A]	T_2BU	1.3-6									
						[J]	TBU	1.1-6									
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0E-7 (ABOUT 99% OF TOTAL FREQUENCY)																	
[B]	RU_R	8.1-7	[C]	SX_S	9.0-7	[C]	TBU	7.6-7		$T_{5,6}BLX$	2.3-7	[B]	AU	4.1-7			
							ATWS	6.9-7									
[B]	TQU_S	7.3-7	[A]	RX_RO	6.0-7	[B]	T_4BU	4.8-7		$T_{5,6}BLX$	1.3-7		I	1.4-7			
			[F]	TQX_S	3.7-7	[H]	$T_{5,6}BU$	2.5-7									
			[E]	TQX_S	2.2-7	[I]	$TB\bar{U}$	1.8-7									
			[A]	T_SQX_S	1.0-7	[D]	$T_{11}BU$	1.8-7									
[C]	SU_S	4.9-7															
[I]	TQU_S	4.8-7															
[C]	V_SU_S	4.2-7															
[F]	TQU_S	4.2-7															
[A]	RV_R	4.1-7															
[D]	T_6QU_S	2.9-7															
Total shown		9.6-6			8.8-6			5.6-5			3.6-7			5.2-6			1.2-5
Other		2.3-7			6.0-8			3.4-7			-			-			5.4-8
Total		9.5-6			8.8-6			5.7-5			3.6-7			5.1-6			1.2-5

Table D.4 OPRA Summary of Core Damage Frequencies for Internal Initiating Events*

Bin I sequences			Bin II sequences			Bin III sequences			Bin IV sequences			Bin V sequences			Bin VI sequences		
Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0 x 10^{-6} (ABOUT 80% OF TOTAL FREQUENCY)																	
						[G]	$T_{12}BU$	1.5-5									
[A]	SY_SX_S	5.0-6				[E]	$T_{10}BU$	4.8-6									
						[F]	T_6BU	4.7-6							[B]	AX_A	4.8-6
															[A]	AX_A	3.3-6
							TWS	2.8-6					TWS	1.7-6			
						[A]	T_2BU	1.2-6				[A]	VR	1.1-6		TWS	1.5-6
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0 x 10^{-7} (ABOUT 95% OF TOTAL FREQUENCY)																	
[B]	RU_R	8.5-7															
			[B]	RX_RO	7.4-7												
			[C]	SX_S	6.9-7												
[B]	TQU_S	5.9-7															
[C]	SU_S	4.4-7	[A]	RX_RO	4.0-7	[C]	TBU	4.2-7									
[A]	RU_R	3.9-7				[B]	T_4BU	4.1-7									
[D]	T_6QU	2.5-7	[F]	$T_{8,13}QX$	2.2-7	[H]	$T_{5,6}BU$	2.2-7									
[I]	T_5QU	2.3-7							[B]	TBLX	1.6-7	[B]	AU	2.3-7			
[E]	$T_{12,14}QU_S$	1.0-7											I	1.4-7			
Total shown		7.9-6			2.1-6			3.0-5			1.6-7			3.0-6			9.6-6
Summation of others not shown		2.0-7			4.0-7			5.0-7			3.0-8			1.0-7			2.0-7
Total core-melt frequency		8.1-6			2.5-6			3.0-5			1.9-7			3.2-6			9.8-6

Notes: 1. The duplications of sequence types within a bin (e.g., two type [A] in bin I) are due to the sequences resulting from steam generator tube ruptures.

2. The sequences are defined by sequence type. This grouping of events considered (1) event-tree sequence type, (2) initiating-event effects if important, and (3) differences that would require unique treatment for the consequence analysis.

3. The TWS sequences are discussed in Appendix E. The line items in this table are summations of all TWS sequences for a bin.

4. Nomenclature: A, large LOCA; T, transient event; S, small LOCA; T_2, loss of main feedwater; T_5, loss of offsite power; T_6, loss of instrument air; T_8, spurious engineered-safeguard actuation; T_{10}, feedline break; T_{12}, loss of low-pressure service water; T_{13}, spurious low-pressurizer-pressure signal; T_{14}, loss of 4-kV switchgear 3TC; R, steam generator tube rupture; VR, reactor-vessel rupture; Q, loss of RCS integrity (transient-induced small-break LOCA); B, failure of RCS heat removal through the steam generators; U_T, failure of core heat removal by HPI cooling; U_S, failure of HPI in injection mode for small LOCAs; U_A, failure of LPI in injection for large LOCAs; U_R, failure of HPI in injection for tube ruptures; $Y_{T,S}$, failure to maintain RCS makeup supply; $X_{T,R,A,S}$, failure to maintain long-term core-heat removal appropriate to the initiating event; L, failure to recover RCS heat removal; O, failure to maintain long-term cooling at hot conditions for tube rupture; I, interfacing-system LOCA.

*Reproduced from OPRA, Table 8.1.

1. INTRODUCTION

1.1 Objective, Scope, and Approach to the Review

Duke Power, in collaboration with the Nuclear Safety Analysis Center, has carried out a full-scope PRA of Oconee Unit 3. The Oconee PRA (OPRA) treats "internally" initiated scenarios (accidents initiated by a functional equipment failure or an external loss of offsite power), as well as externally initiated scenarios (e.g., earthquakes, floods, etc.) and other physical phenomena (fires, internal floods). Containment analysis was also performed in the PRA. BNL has conducted a full-scope review of the "internal" and "external event" scenarios defined in the OPRA. However, only a limited review of the containment performance and radiological source term analyses in the OPRA has been performed. The present volume describes the review of internal events scenarios out to core damage; the review of external events is presented in Volume 2 of this report. The review of the OPRA containment performance and radiological source terms will be presented in a separate report.

The broad objective of the BNL review of the internal events portion of the OPRA was to evaluate qualitatively and quantitatively the study's assessment of the important accident sequences that are internally generated and lead to core damage. In addition, the review included an assessment of the externally generated LOOP accident initiator. To carry out this objective, BNL reviewed the assumptions and methods of the OPRA within its stated scope. Within this scope and within the basic methodological framework of the OPRA, BNL reevaluated the important accident sequences that lead to core damage, their respective frequencies of occurrence, the total frequency of core damage, and the associated uncertainties. The review included evaluations of accident initiators, data, and accident sequence development and quantification. In addition, a limited ATWS sensitivity analysis was performed.

The internally generated accident initiators reported in this NUREG/CR were reviewed over a seven-month period by five people at BNL (three on a regular basis and two on a partial basis). D. Xue, D. Ilberg, and N. A. Hanan of the Risk Evaluation Group were the main project engineers and they reviewed the frequencies of the accident initiators, the accident sequence and system modeling, the data base, the quantification of the event and fault trees, and they also performed the uncertainty analyses. R. G. Fitzpatrick and T-L. Chu contributed to the qualitative review of systems fault trees and the interfacing system LOCA.

The project monitor was E. Chelliah of the Reliability and Risk Assessment Branch, Division of Safety Review and Oversight, U.S. Nuclear Regulatory Commission.

The review benefited from a productive meeting and a plant visit held between NRC, BNL, and DPC. DPC staff provided the information and discussion needed to gain a detailed understanding of the PRA for the in-depth review process. The various submittals, with which DPC responded to the various BNL comments in the meeting and plant visit, always constituted a technical improvement to the PRA and were always responsive to the BNL comments.

1.2 Organization of Report

Section 2 gives a description of the plant modeling which includes identification of initiating events that can lead to core damage and a discussion of safety functions and systems important to the prevention of core damage events. Section 3 presents a description of accident sequence definition and a discussion of the event tree/fault tree approach used in the OPRA. Section 4 presents the quantification of the initiating events' frequencies and the evaluation of the data base used in the OPRA and in this review. Section 5 provides accident sequence quantification, reviews the numerical values of the parameters necessary for this quantification, gives a brief description of the OPRA approach to quantification, and presents the BNL comments on the quantification process and the revised core damage frequencies. This section also contains an analysis of the uncertainties in the core damage frequency and a limited ATWS sensitivity analysis. Details of the BNL core damage accident sequences are given in the Appendices.

2. PLANT MODELING

This section reviews the modeling of the plant in the Oconee PRA[1] (OPRA). The plant modeling includes the identification of

- Safety functions important to prevent or mitigate core damage.
- Systems that directly perform these safety functions (frontline systems).
- Systems that support the frontline systems (support systems).
- Success criteria of the safety functions and systems.
- Initiating events that can lead to core damage.

Subsection 2.1 describes the safety functions, the corresponding frontline and support systems, and their success criteria. Subsection 2.2 presents the initiating events and their grouping according to the success criteria for the frontline systems. In both subsections, the Oconee PRA assumptions are reviewed, evaluated, and compared to those of the Oconee RSSMAP study,[2] the Arkansas PRA,[3] and the Midland PRA.[4]

2.1 Safety Functions and Corresponding Systems

2.1.1 Safety Functions and Frontline Systems

The safety functions important to preventing or mitigating the consequences of core damage following any initiating event at Oconee-3 are given in Table 2.1. Each function in this table is directly performed by one or more frontline systems. The Oconee-3 frontline systems performing each of these functions for a transient and for a LOCA are given in Tables 2.2 and 2.3, respectively. A comparison of the frontline systems for Oconee-3, Arkansas-1, and Midland-2 is presented in Table 2.4. Note from this table that the frontline systems for these three plants are almost identical. A detailed description of the frontline systems operations and response to transients and LOCAs for Oconee-3 can be found in Appendix A of the OPRA,[1] and in the FSAR.[5]

2.1.2 Success Criteria for the Frontline Systems

The Oconee PRA considers four general classes of initiating events:

1. Transients with successful scram.
2. Anticipated transients without scram (ATWS).
3. Loss-of-coolant accidents (LOCAs).
4. Steam generator tube rupture.

Even though not specifically stated in the Oconee PRA, the success criteria for the prevention of core damage are defined in terms of the minimum number of systems required to prevent core uncovery or excessive fuel clad temperature. In the case of ATWS, it is stated that incipient core melt is used for the definition of the success criteria. This definition is said to be based on realistic (best estimate) predictions.

The following subsections discuss the success criteria used in the Oconee PRA and compare them with those used in the Oconee RSSMAP[2] study, the Arkansas PRA,[3] and the Midland PRA.[4] The following subsections also present the success criteria used in this BNL review whenever they differ from those of the OPRA.

2.1.2.1 Success Criteria for Transient Initiators

The success criteria used in the OPRA for transient initiators are summarized in Table 2.5. These criteria were reviewed and considered reasonable. Note that except for the "Feed and Bleed" success criteria (high pressure injection with relief through the PORV or SRVs), all the other criteria are the same as those used in the Oconee FSAR.[5] In Tables 2.6 through 2.8, the success criteria used in the Oconee RSSMAP,[2] the Arkansas PRA,[3] and the Midland PRA[4] are presented, and it can be seen that they are very similar to the ones used in the Oconee PRA.

2.1.2.2 Success Criteria for ATWS Initiators

The success criteria for ATWS initiators used in the OPRA are based on analysis performed by B&W.[6,7] These success criteria, and those of the Oconee RSSMAP, Arkansas, and Midland, are also presented in Tables 2.5 through 2.8. From these tables the following conclusions can be made:

- The success criteria used in the OPRA and the MPRA are very similar, as are those used in the Oconee RSSMAP and the Arkansas PRA.
- The two sets of success criteria differ primarily in the assumption by the OPRA and the MPRA that a total loss of feedwater or failure of pressure relief will endanger the reactor integrity and perhaps result in a LOCA which can be mitigated by HPI and/or LPI.

This BNL review is in agreement with the success criteria used by the OPRA.

2.1.2.3 LOCA Success Criteria

The determination of the LOCA success criteria is briefly discussed. The OPRA approach, comparison with some other PRAs, and the BNL review are given in the following paragraphs.

The OPRA Approach

The OPRA provides success criteria for the small- and large-break LOCAs. Separate success criteria for the injection and recirculation phases are given. The derivation is based on several sources including the RSSMAP,[2] the FSAR,[5] and best-estimate calculations.

The OPRA provides the following explanations based on physical phenomena in establishing its success criteria:

1. For large LOCAs, depressurization actuates the LPI and the CFT. The BWST may be depleted in about 30 minutes by the LPI injection, requiring a manual transfer to the LPR mode. One CFT is needed for large LOCA to prevent excessive peak cladding temperatures and to provide injection flow

during the early stages for the breaks at the lower end of the spectrum (4 to 10 inches). Breaks larger than 4 inches quickly depressurize the RPV to the point at which substantial LPI flow can be delivered. (This break size was chosen as the boundary between small- and large-break LOCAs.)

2. For small LOCAs, when the RBSS is actuated (because of loss of RBCS or when the reactor building pressure reaches the 10 psi set point), the BWST will be depleted in about two hours. Otherwise, it will suffice for HPI injection for at least 12 hours. When low-low level is reached in the BWST, assumed to occur at either two or twelve hours, manual operator action to align the HPR or the LPR is required.

3. For small LOCAs, there are two ranges of break sizes for which RPV pressure behaves differently if heat removal by the steam generators is lost. For breaks smaller than about 1.5 in., the RCS may not depressurize since the break alone provides insufficient capacity to remove decay heat. The RCS will tend to pressurize to the point at which the pressurizer PORV and/or SRVs cycle. It is stated in the OPRA that calculations indicate that the PORV/SRV discharge to containment will be sufficient to cause HPI automatic actuation at the 3 psi set point within one hour; this is the time at which core uncovery starts. Thus, the OPRA treats the two subgroups in the same way, as a simple group of small LOCAs.

The OPRA success criteria are summarized in Table 2.9. Note that a successful high pressure recirculation (HPR) consists of one of three HPI pumps, one of two LPI pumps, and one of two decay heat removal heat exchangers.

Comparison With Other PRAs

Table 2.9 compares OPRA success criteria with other PRAs; this table also presents the success criteria used in this review. It can be seen that either a 4-in. diameter or a 0.1-ft^2 break criterion (the two are very similar) was used as a boundary between small- and large-break LOCA in all PRAs. For the small LOCA range, there is almost total agreement on success criteria, the only difference being the case of an open SRV in the Arkansas IREP in which two HPI pumps were assumed to be necessary. The Midland PRA[4] includes correspondence with B&W, in which B&W recommends the use of two HPI pumps for about 20 to 30 minutes, while later one HPI pump is stated to be sufficient. Since core damage with no HPI pump working is not initiated in less than 30 to 40 minutes, and no reason for the 2/3 HPI criterion is given in the IREP[3] study, the validity of this criterion is difficult to judge.

For large LOCAs, two main differences can be noted:

a. The requirement for a HPI pump in addition to LPI in some PRAs in the injection phase for breaks in the 4- to 10-in. range.

b. The number of CFTs considered to be required.

OPRA has acknowledged the possibility of using HPI for the 4- to 10-in. break, but determined that rapid depressurization to the point of substantial LPI flow will occur (see item 1 in the preceding section). The correspondence with B&W included in the Midland PRA[4] also touches on this and points out that

LOCA analysis has not routinely explored all RCS pipe breaks outside the design basis conditions. In its response, B&W states that HPI pumps are necessary for smaller breaks where the pressure remains high for a longer time, and LPI pumps are necessary for large breaks where large amounts of water must be supplied to balance the rapid rate at which it is being lost. This indicates that <u>either</u> HPI <u>or</u> LPI is generally needed, rather than <u>both</u> systems. Some old physical calculations (BAW-10052) show that a rapid pressure decrease (in less than 5 minutes) to below 200 psi is obtained for 0.5- and 0.3-ft^2 breaks. Thus, uncertainty with respect to the success criteria seems limited to a narrow range of 0.1 to 0.2 ft^2. This is smaller than the 0.1- to 0.6-ft^2 range in which this success criterion is applied in the other PRAs. BNL judges that the Oconee PRA criteria for the entire range of large LOCAs are more realistic. Notwithstanding, the results of this review indicate that the effect of using a requirement for both HPI and LPI is very small.

The second difference among the PRAs is the number of CFTs required. The main reason CFTs are needed is to limit peak cladding temperatures (see item 1 in the preceding section). In the B&W response to the MPRA,[4] it is stated that massive core damage would not be caused by CFT failure to operate, nor in general, would CFTs substitute for the required LPI or HPI. On the basis of the above, BNL determined that the OPRA CFT success criterion is realistic and used the same criterion without change. The BNL evaluation of the large LOCA accident sequences (Section 5) revealed that the effect of requiring two CFTs rather than one on core damage frequency is very small.

2.1.2.4 Steam Generator Tube Ruptures Success Criteria

The SGTR is a special case of a very-small-break LOCA, i.e., the rupture causes a slow decrease in RCS pressure. The success criterion for the injection phase is one out of three HPIs, as in the small-LOCA case. Manual actuation of the HPI is considered necessary for the SGTR if feedwater to SGs is not available, and this is further discussed in the subsection on LOCA system response in Chapter 3.

The main difference between SGTR and a very small LOCA appears in the definition of the end states. For a SGTR, the most desirable end state is to achieve long-term cooling at cold shutdown; in this case, the success is established by achieving the entry conditions for Decay Heat Removal (DHR) and having the DHR system working (one out of two LPI trains including the heat exchanger, with suction from the RCS). If that fails, a long-term cooling at hot shutdown conditions can still be maintained by replenishing the BWST (success of the HPIS is implicit), or by using the HPR with the isolation of the affected steam generator. Table 2.10 compares the long-term cooling success criteria for small LOCAs and SGTR.

2.1.3 Support Systems

This section briefly discusses the major systems supporting the frontline systems in Oconee-3 and compares these systems with the corresponding support systems in the Arkansas Nuclear One and the Midland-2 plants. A summary of these systems for all three plants is presented in Table 2.11.

Electric Power System (ac and dc). The major characteristics of the Oconee-3, Arkansas, and Midland-2 electric power systems are summarized in Table 2.11. The most significant differences are:

- Oconee-3 has three ac load divisions, while Arkansas and Midland have only two.

- Oconee-3 uses the two Keowee hydroelectric generators and the 100-kV transmission system from the Lee Steam Station as the onsite power. Arkansas and Midland have two 4160-V emergency diesel generators.

Service Water System. The service water system [called low pressure service water system (LPSW) in Oconee-3] is designed to remove heat from plant auxiliaries which are required for a safe shutdown. Table 2.11 summarizes the features of this system for Oconee, Arkansas, and Midland. A comparison of the most important loads supported by the service water system is given below.

- For all three plants, the service water system is used by the reactor building cooling system, the HVAC room cooling, and the component cooling system.

- For Oconee and Arkansas, the service water system is directly used for cooling the motor of the HPI pumps (this cooling has an automatic backup from the high pressure service water system in Oconee).

- For Arkansas and Midland, the service water system is used for cooling the diesel generators.

- For Oconee, the service water system is directly used for cooling the emergency feedwater system (cooling of the steam-driven emergency feedwater pump is automatically backed up by the high pressure service water system). In Midland, only the motor-driven EFW pump is directly cooled by the service water system.

- In Arkansas, the low pressure injection and the spray pumps are cooled by the service water system. In Oconee, these pumps are only dependent on the service water system through the HVAC system. In a meeting with DPC, it was stated that the HVAC room cooling is no longer required for these pumps, since calculations show that they can operate for days without any room cooling; this consideration is included in the final analysis for the OPRA and for this review. In Midland, these pumps are cooled by the component cooling system.

- In Oconee, the decay heat coolers (decay heat exchangers) use the service water system for RCS cooling.

- In Midland, the service water system is shared by units 1 and 2.

It is important to note that the Oconee-3 service water system has interties with the same system on the other two units (Oconee-1 and -2), and also with the high pressure service water which is common for all three Oconee units.

Heating, Ventilation, Air Conditioning System (HVAC). This system is similar for all three plants. In Arkansas and Midland, it is stated that the HVAC is required for ac and dc switchgear rooms. The Oconee PRA does not use this as a requirement, nor does the Oconee FSAR. Since the OPRA and the Oconee FSAR were the main sources for the BNL review, this assumption was also used in this report. Table 2.11 summarizes the important loads for this system.

Instrument Air System. In the Arkansas PRA, it is claimed that several non-safety systems require air for proper operation and that safety-related components fail safe upon loss of air. In the Midland PRA, no analysis of the instrument air system is presented; however, the Midland FSAR makes the same claims as the Arkansas PRA.

In Oconee, the instrument air and the station air systems are considered as one system and referred to as the Compressed Air System. This system is very important for the proper operation of the MFW and EFW systems and the reactor coolant makeup system, and for the control of the flow rate of the service water used in the decay heat coolers. It is also important because much of the instrumentation depends upon the compressed air system. The importance of the compressed air system can be seen in the dominant core damage sequences presented in Section 5 and Appendix A of this report.

Engineered Safeguards Actuation System. This system is very similar for all three plants and a summary is given in Table 2.11.

Integrated Control System. This system's most important function is proper coordination between reactor, steam generators, main feedwater, and turbine during normal operation. Even though a complete (detailed) description of this system is not given in any of the FSARs or PRAs (the Oconee PRA is by far the best description), its major features are similar for the three plants. For Oconee and Arkansas, the present ICS design essentially eliminated ICS-caused failures of safety systems.

Component Cooling Water System. In Oconee-3, the only important function of the component cooling system is the cooling of the thermal barrier of the reactor coolant pump seals. In Midland, the component cooling system is more important than in Oconee because it is required by the following loads: decay heat removal, heat exchangers, high pressure injection pumps seal coolers, low pressure injection pumps seal coolers, the reactor building spray pumps and coolers, and the RCP seal coolers.

In the Arkansas PRA, the component cooling water system is not mentioned.

2.1.4 Frontline and Support Systems Dependences

In Sections 2.1.1 and 2.1.3, the frontline and major support systems were briefly described. Major differences in the loads served by the support systems were also presented in the preceding section. To give a better perspective of the dependences among frontline vs support systems and support vs support systems in Oconee-3, Tables 2.12 and 2.13 present dependence tables to illustrate the functional dependences of these systems.

2.2 Initiating Events

This discussion of the selection of initiating events that could challenge the safety systems is divided into three parts. The first describes the approach used in the OPRA,[1] the second compares this with the Midland,[4] Arkansas,[3] and RSS-PWR[2] approaches, and the third presents the results of the BNL review with respect to the choice of initiating events.

The OPRA considers the following general classes of initiating events:

a. Loss-of-coolant accidents (LOCAs).
b. Transients with successful scram (T_i).
c. Anticipated transients without scram (ATWS).
d. External events (analyzed in a separate report).

2.2.1 OPRA Initiators Selection

2.2.1.1 LOCA Initiators

The LOCA initiators are subdivided into two groups according to the size of the break and the corresponding frontline system response:

a. Large LOCAs - equivalent break size diameter about 4 in. or more, for liquid or steam breaks.

b. Small LOCAs - equivalent break size diameter about 4 in. or less, for liquid or steam breaks.

In addition, three other cases can be considered as belonging to these classes of initiating events:

a. Steam Generator Tube Rupture (SGTR). A 400-gpm flow rate from primary to secondary is considered representative of this initiator in the OPRA.

b. Reactor Pressure Vessel (RPV) rupture. OPRA assumes that RPV ruptures would be beyond the capability of engineered safety features.

c. Interfacing-System LOCA.

BNL considers this selection of LOCA initiators to be acceptable. BNL's only change was to divide the OPRA small LOCAs into two subgroups:

a. Very small LOCAs - equivalent break size diameter about 1.5 in. or less.

b. Small LOCAs - equivalent break size diameter between 1.5 and 4 in.

The advantage of this division is that it is more representative of the initiating-event frequencies for these break sizes as discussed in Section 3, and it allows for a better treatment of the automatic actuation of RBSS when the containment reaches 10 psi; in the upper-range break sizes (1.5 to 4 in.), the RBSS is automatically actuated even with the RBCS in operation. This is believed to be more realistic.

2.2.1.2 Transients With Successful Scram

The transient initiators for which scram is successful are divided into 14 different groups, imposing the same success requirement on the frontline system. They are characterized in the OPRA Table 3.5:

T_1	Reactor/turbine trip
T_2	Loss of main feedwater
T_3	Partial loss of main feedwater
T_4	Loss of condenser vacuum
T_{5SUBF}	Failure of offsite power due to 230-kV substation failure
T_{5FEEDF}	Failure of offsite power due to grid or feeder failure
T_6	Loss of air
T_7	Excessive feedwater
T_8	Spurious engineered safeguards actuation signal
T_9	Steam-line break
T_{10}	Feedwater-line break
T_{11}	Loss of ICS power bus 3KI
T_{12}	Loss of low pressure service water
$T_{12(108)}$	Loss of low pressure service water due to transfer closed of valve LPSW-108
T_{13}	Spurious low pressurizer pressure signal
T_{14}	Loss of 4-kV switchgear 3TC

The list of transient initiators used in OPRA was obtained by a systematic evaluation of several sources, of which the most important are the following:

a. An EPRI Survey[8] of operating experience with PWRs in which 41 transient initiators are identified. Table 2.14 lists the grouping of some of these initiators into OPRA transient initiators' groups.

b. Feedback from the plant system review of all the safety-related and non-safety-related systems with special emphasis on support systems that could affect the frontline systems. Also, feedback from the event tree construction phase was incorporated.

c. The Oconee plant-specific experience.

d. The Reactor Safety Study[10] -- Table I.4-9 for PWR transients.

The review of the above sources, combined with the experience of the PRA team, resulted in a master list of 44 initiators which are shown in Table 3.3 of the OPRA. The grouping of these into 5 LOCA initiators and 14 transient initiators is given in Table 3.4 of OPRA. BNL considers this list and the qualitative grouping into the OPRA 14 transients to be correct; this is further shown in Section 2.2.3.

2.2.1.3 ATWS: Anticipated Transient Without Scram

If the reactor protection system fails to scram the reactor after any of the transient initiating events, then an ATWS results. OPRA combined the 41 transients of the EPRI report[8] into the following 12 groups of ATWS initiators which correspond to those addressed by NRC[9] and B&W[6] reports:

1. Loss of condenser vacuum.
2. Turbine trip (TT).
3. Loss of main feedwater (LMFW).
4. Loss of offsite power (LOOP).
5. Load increase.
6. Loss of RCS flow.
7. Control rod withdrawal (CRW).
8. RCS depressurization.
9. Boron dilution.
10. Excessive cooldown.
11. MSIV closure.
12. Inactive RCS loop startup.

The collapse of the 41 EPRI initiators into this scheme is not provided in the PRA nor is the relation between the 14 initiators used as initiators of the transients with scram and the 12 ATWS listed here. BNL has reproduced this relationship in order to review the quantification of the ATWS initiators. The grouping believed to be used in OPRA ATWS is shown in Section 4 where the BNL quantification of ATWS initiators is discussed.

OPRA has removed 8 of the above ATWS initiators, for the following reasons:

a. Numbers 11 and 12 are not directly applicable to Oconee-3.

b. Numbers 5 and 10 are insignificant contributors because of frequency smaller than 0.01 per year in B&W plants, and are bounded by LOOP and LMFW.

c. Numbers 7 and 9 again have low frequency and are bounded by other transients (TT).

d. Number 6 is not included in the ATWS analysis because it is accounted for in the LOOP event.

e. Number 8 is not included in the ATWS analysis because it is included in the case of a small LOCA or a stuck-open relief valve that causes the plant to scram on low RCS pressure. Since the failure of RPS occurs at low RCS pressure, the ensuing pressure surges will be smaller than for the other ATWS initiators. Furthermore, the loss of water inventory has the effect of creating a higher void fraction in the core that reduces the core power. Considering the less severe system response and the lower frequency of LOCA, it is not considered to be a significant ATWS contributor compared to the main four that are finally considered.

Thus, OPRA considers only four ATWS initiators for the accident sequence evaluation:

1. Loss of condenser vacuum.
2. Turbine trip.
3. Loss of main feedwater.
4. Loss of offsite power.

The OPRA ATWS analysis is a scoping analysis, and sequences that may contribute to core damage, but which have small impact, were not further addressed. On this basis, the choice of the above four initiators is considered appropriate.

The collapsing of the OPRA 12 transient initiators to the four ATWS initiators is not given in the OPRA. It is this review's assessment that the turbine trip group includes transients T_1, T_3, T_7, T_8, T_9, T_{11}, and T_{13}; the loss of condenser vacuum group includes T_4; the loss of power group includes T_5; and the loss of main feedwater group includes T_2 and T_6.

The OPRA states that the ATWS scoping analysis has considered the initiators in a conservative way because of the grouping, and the use of all power levels (0-100% power). BNL agrees that the consideration of transients with power less than 25% is conservative, but believes that the overall grouping of initiators disregarding power level is a realistic representation of ATWS initiators.

The quantification of the ATWS initiators is reviewed in Section 4.

2.2.1.4 External Events

OPRA considers several external events in different depth and detail. Most detailed analysis is provided for internal floods and earthquakes. Other external events treated are fires, tornados, external floodings, and airplane crashes. The external events are considered in a separate report.

2.2.2 Comparison with Arkansas IREP[3] and Midland PRA[4]

The Arkansas and Midland PRAs have systematically generated their initiating event lists in an effort to make them as complete as possible. Arkansas IREP claims to have performed a failure mode and effect analysis (FMEA) on the RCS piping and the frontline systems as well as their support systems. The RCS piping FMEA considered all break sizes and locations. The FMEA performed for transient initiators' identification considered frontline and support system response or their availability after a single fault in any of these systems.

The Arkansas analysis and the grouping of the initiating events resulted in eight groups of transient initiators and six groups of LOCA initiators. In addition, interfacing LOCA was also considered, although it was later disregarded because of very low probability of occurrence. The LOCA initiators for Arkansas are shown in Table 2.9. The use of more LOCA initiators is based on the different success criteria used, which were discussed earlier in Subsection 2.1.2.3. The eight transient initiators for Arkansas are also given in Table 2.11.

The Midland PRA similarly used a systematic and detailed approach to generate its initial list of 47 initiators. It constructed a master logic diagram searching for equipment malfunction that causes excessive offsite releases by degrading the main safety functions. The list of 47 initiators was further grouped into 25 initiators consisting of 6 LOCA initiators (including SGTR and interfacing LOCA), 10 transients, and 4 external event initiators. The 10 Midland transient initiators are shown in Table 2.14 where they are compared with Arkansas and OPRA. The table shows the grouping of some of the transients in EPRI-NP-2230[8] into the groups of the PRAs considered.

2.2.3 BNL Assessment of the Selection of Initiating Events

The three PRAs considered in the preceding sections have all used a systematic approach to generate a master list of initiating events and have grouped them into their final list of initiators which are considered to generate similar system response and have similar success criteria.

The LOCA subdivision was considered earlier in the discussion of LOCA success criteria where it was shown that on the basis of the more realistic success criteria used in the OPRA the large- and small-LOCA groups can represent the entire spectrum of breaks. The only change considered to provide a more realistic analysis is the division of the "S" group used in the OPRA into very small LOCAs, VS (<1.5 in.), and small LOCAs, S (1.5 to 4 in.).

Other LOCAs, e.g., interfacing LOCAs, are considered in all three PRAs. SGTR is considered only in Midland and OPRA.

The transient initiator selection is compared in Table 2.14. It can be seen that OPRA accounted for all transient initiators considered in the other PRAs. Loss of air, which is considered separately in OPRA, was grouped in the Midland PRA into the loss of FW transient. It should be considered separately in OPRA because it also affects the availability of the emergency feedwater system. Loss of dc power, considered in Arkansas IREP, was also considered in OPRA in a special study of the electrical power supply system, and it was determined to be a minor contributor because of the high redundancy in the power system. This special study identified only T_{14} as a significant initiator. Thus, BNL concludes that the list of 14 transient initiators in OPRA is acceptable.

It can be seen from Table 2.14 that OPRA has used, in some cases, a more refined grouping of transients. The loss of condenser vacuum which was treated separately in OPRA was included in loss of main feedwater in Arkansas and Midland PRAs. Loss of condenser vacuum in OPRA was considered to have lower (less successful) probabilities of recoveries than in the loss-of-main-feedwater transient sequences. The partial loss of feedwater which was separated out in the case of OPRA was treated differently in the other two PRAs; this can be recognized if it is noted that EPRI transients Nos. 15 and 22 are the main contributors to this group. Midland PRA includes both in the reactor trip case which does not affect any front system availability; Arkansas IREP includes part as turbine trip (No. 15) and part as the more severe transient of loss of main feedwater (No. 22). OPRA acknowledges the unavailability of part of the main feedwater system, and consequently its treatment of these initiators is more realistic.

2.3 References

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2. Kolb, G. J. et al., Reactor Safety Study Methodology Applications Program: Oconee #3 PWR Power Plant, NUREG/CR-1659 (Vol. 2), May 1981.

3. Kolb, G. J. et al., Interior Reliability Evaluation Program: Analysis of the Arkansas Nuclear One-Unit 1 Nuclear Power Plant, NUREG/CR-2787, June 1982.

4. Midland Nuclear Plant - Probabilistic Risk Assessment, Consumers Power Company and PLG Inc., May 1984.

5. Oconee Final Safety Analysis Report - Duke Power Company.

6. Analysis of B&W NSS Response to ATWS Events, BAW-1610, June 1980.

7. McBride, A. F. et al., Babcock & Wilcox Anticipated Transients Without Scram Analysis, BAW-10099, Rev. 1, May 1977.

8. McClymont, A. S. and Poehlonan, B. W., ATWS: A Reappraisal - Part 3: Frequency of Anticipated Transients, EPRI NP-2230, Jan. 1982.

9. Anticipated Transients Without Scram for Light Water Reactors, NUREG-0460, Nov. 1981.

10. Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, NUREG/75-014, 1975.

Table 2.1 Safety Functions for Oconee-3

1.	Reactor subcriticality
2.	Reactor coolant system (RCS) overpressure protection/RCS integrity
3.	Core cooling
4.	RCS inventory makeup
5.	Containment overpressure protection
6.	Radioactivity removal

Table 2.2 Oconee-3 Transient Safety Functions/Frontline Systems

Safety Function	Frontline System(s)
Reactor subcriticality	a. Reactor protection system (RPS) b. High pressure injection system (HPIS)
Reactor coolant system (RCS) Overpressure protection	a. PORV and SRVs
Core cooling	a. Power conversion system (PCS) b. Emergency feedwater system (EFWS) c. High pressure onjection system (HPIS) and pilot-operated relief valve (PORV) or safety relief valves (SRVs)
RCS invertory makeup	a. HPIS
Containment overpressure protection	a. Reactor building cooling system (RBCS) b. Reactor building spray system (RBSS)
Radioactivity removal	a. RBSS

Table 2.3 Oconee-3 LOCA Safety Functions/Frontline Systems

Safety Function	Frontline System(s)
Reactor subcriticality	a. RPS
Core cooling	
• Injection phase	a. HPIS b. LPIS c. Core flooding system (CFS) d. EFWS e. PORV and SRVs
• Recirculation phase	a. High pressure recirculation system (HPRS) b. Low pressure recirculation system (LPRS) c. Decay heat removal system (DHRS)
Containment overpressure protection	
• Injection phase	a. RBCS b. RBSS - injection mode
• Recirculation phase	a. RBCS b. RBSS - recirculation mode
Radioactivity removal	a. RBSS - injection and recirculation modes

Table 2.4 Comparison of Oconee-3, ANO-1, and Midland-2 Frontline Systems

	Oconee-3	ANO-1	Midland-2
Reactor vendor	B&W	B&W	B&W
Power (MWe)	886	886	886
Containment	Dry	Dry	Dry
No. of PORV	One opening at 2450 psig	One opening at 2450 psig	One opening at 2260 psig
No. of SRVs	Two openings at 2500 psig	Two openings at 2500 psig	Two openings at 2500 psig
High pressure injection system	3 pumps (2900-psig shutoff head) Injects into 4 RCS cold legs Actuates upon RCS pressure of 1550 psig or containment pressure of 3 psig	3 pumps (2900-psig shutoff head) Injects into 4 RCS cold legs Actuates upon RCS pressure of 1500 psig or containment pressure of 4 psig	3 pumps (3200-psig shutoff head) Injects into 4 RCS cold legs Actuates upon RCS pressure of 1500 psig or containment pressure of 4 psig
Low pressure injection system	2 pumps (a third pump is available; it is normally valved out and is load shed) Injects into reactor vessel via 2 low pressure injection lines Actuates upon RCS pressure of 550 psig or containment pressure of 3 psig	2 pumps (190-psig shutoff head) Injects into reactor vessel via 2 low pressure injection lines Actuates upon RCS pressure of 1500 psig or containment pressure of 4 psig	2 pumps (200 psig shutoff head) Injection into reactor vessel via 2 low pressure injection lines Actuates upon RCS pressure of 1500 psig or containment pressure of 4 psig
Core flooding system	2 tanks Injects into reactor vessel via 2 low pressure injection lines Actuation upon RCS pressure of 600 psi	2 tanks Injects into reactor vessel via 2 low pressure injection lines Actuation upon RCS pressure of 600 psi	2 tanks Injects into reactor vessel via 2 independent injection lines Actuation upon RCS pressure of 600 psi

Table 2.4 Continued

	Oconee-3	ANO-1	Midland-2
Reactor building cooling system	3 containment fan coolers	4 containment fan coolers	4 containment fan coolers
	Actuates upon containment pressure of 3 psig	Actuates upon containment pressure of 4 psig	Actuates upon containment pressure of 4 psig or RCS pressure of 1500 psig
Reactor building spray system (containment spray system in Midland)	2 pump trains	2 pump trains	2 pump trains
	Sprays containment atmosphere via 2 spray headers	Sprays containment atmosphere via 2 spray headers	Sprays containment atmosphere via 2 spray headers
	Actuates upon containment pressure of 10 psig	Actuates upon containment pressure of 30 psig	Actuates upon containment pressure of 30 psig
Emergency feedwater system	3 pumps (2 electric, 1 turbine)	2 pumps (1 electric, 1 turbine)	3 pumps (2 electric,* 1 turbine, modified as per PRA)
	Injects into both once-through steam generators	Injects into both once through steam generators	Injects into both once-through steam generators
	Actuates on both main feed pumps trip, or low pressure at the discharge of both FW pumps	Actuates on reactor coolant pump trip, main feed pump trip, low steam generator level, low steam generator pressure	Actuates on 3 out of 4 reactor coolant pump trip, both main feed pumps trip, low steam generator pressure, emergency core cooling actuation system
Power conversion system	3 electric condensate, 2 steam main feed	3 electric condensate, 2 steam main feed, 1 auxiliary feed	2 electric condensate, 2 steam main feed
	Normal post-trip steam generator cooling system	Normal post-trip steam generator cooling system	Normal post-trip steam generator cooling system

*The second electric pump was not considered in the definition of success criteria (see Table 2.8).

Table 2.5 Transient Success Criteria for Oconee

Subcriticality	Core Cooling	Reactor Coolant System (RCS) Overpressure Protection	RCS Integrity	RCS Inventory Makeup	Containment Overpressure Protection Due to Steam Evolution	Post-Accident Radioactivity Removal
>6 Control rod groups inserted into core by the reactor protection system (RPS)	Given RPS success* power conversion system (PCS) OR 1/3 Emergency feedwater system (EFS) to one SG OR 1/3 High pressure injection system (HPIS) and PORV valve opens OR 2/3 HPI pumps and SRVs open Given RPS failure PCS and HPIS (1/3) OR 1/2 MD EFW pump, the TD EFW pump and one HPI pump OR Borated water injection with HPI/LPI, given a RCS break.	Given RPS Success 1/2 Safety Relief Valves OR PORV open when demanded Given RPS Failure PORV and 1/2 SRVs open when demanded	All safety/PORV relief valves reset after opening	1/3 HPIS	1/3 Reactor building cooling system fan coolers OR 1/2 Reactor building spray injection system	1/2 Reactor building spray injection system

*The standby shutdown facility auxiliary service water system is also a successful means of core cooling.

Table 2.6 Transient Success Criteria for Oconee-3 (RSSMAP)

Subcriticality	Core Cooling	Reactor Coolant System (RCS) Overpressure Protection	RCS Integrity	Containment Overpressure Protection	Post-Accident Radioactivity Removal
>6 Control rod groups inserted into core by the reactor protection system	Power conversion system OR 1/3 Emergency feed-water system OR High head auxiliary service water system OR 1/3 High pressure injection system	1/3 Safety/relief valves open when demanded	All safety/relief valves reset	1/3 Reactor building cooling system fan trains OR 1/2 containment spray system w/recirculation	1/2 containment spray system

Table 2.7 Transient Success Criteria for Arkansas Nuclear One

Subcriticality	Core Cooling	Reactor Coolant System (RCS) Overpressure Protection	RCS Integrity	RCS Inventory Makeup	Containment Overpressure Protection
>6 Control rod groups inserted into core by the reactor protection system (RPS)	Given RPS success Power conversion system (PCS) OR 1/2 Emergency feed-water system (EFS) OR 1/3 High pressure injection system (HPIS) and 1/3 safety/PORV valves open	Given RPS success 1/2 safety relief valves open when demanded Given RPS failure 2/2 safety relief valves open	All safety/PORV relief valves reset after opening	1/3 HPIS	1/4 Reactor building cooling system fan coolers OR 1/2 Reactor building spray injection system
	Give: RPS failure PCS and HPIS and 2/2 safety relief valves open OR EFS and HPIS and 2/2 safety relief valves open				

*From Table A.4 in NUREG/CR-2787.

Table 2.8 Transient Success Criteria for Midland-2

Subcriticality	Core Cooling	Reactor Coolant System (RCS) Overpressure Protection	RCS Integrity	RCS Inventory Makeup	Containment Overpressure Protection Due to Steam Evolution	Post-Accident Radioactivity Removal
59 out of 61 control rods	<u>Given RPS success</u> power conversion system (PCS) <u>OR</u> 1/2 Auxiliary feed-water system (AFWS) <u>OR</u> One high pressure injection system pump (HPIS) and 1/3 safety/PORV valves open <u>Given RPS failure</u> PCS and HPIS (1/2) <u>OR</u> 1/2 AFW and HPIS (1/2) <u>OR</u> Reactor vessel, head lift, and one HPI and one LPI	<u>Given RPS success</u> 1/2 Safety relief valves <u>OR</u> PORV open when demanded <u>Given RPS failure</u> • With turbine trip • 2/3 Relief valves (PORV/SRVs) open • Without turbine trip • 3/3 Relief valves open <u>OR</u> • Reactor vessel head lift	All safety/PORV relief valves reset after opening	One HPIS pump	1/3 Reactor building cooling system fan coolers <u>OR</u> 1/2 Containment spray system	1/2 Reactor building spray injection system

Table 2.9 A Comparison of OPRA LOCA Success Criteria With Other PRAs and the BNL Review[1]

Break Size		Oconee PRA		Oconee RSSMAP		Midland PRA		Arkansas IREP		BNL Review	
Dia. [inch]	Area [ft^2]	Injection Phase	Recirculation Phase	Injection Phase	Recirculation Phase	Injection Phase	Recirculation Phase	Injection Phase	Recirculation Phase	Injection Phase	Recirculation Phase
0.5 1.0 1.5	0.001 0.003 0.01	1/3 HPI	1/3 HPR or [1/2 LPR and cooldown] if BWS; suffice 12 hr	1/3 HPI	1/3 HPR	1/2 HPI	1/2 HPR	1/3 HPI	1/3 HPR or [1/2 EFS during injection and 1/2 DHRS]	1/3 HPI	Same as Oconee PRA
	0.03		1/3 HPR if BWST suffice 2 hr			Same as above	Same as above	2/3 HPI	1/3 HPR		
								1/3 HPI	Same as above	Same as above[3]	Same as above[3]
4.0	0.1 0.3	1/2 LPI and 1/2 CFT	1/2 LPR at 30 min when BWST depleted	1/3 HPI and 1/2 LPI[2]	1/2 LPR	1/2 HPI and [1/2 LPI or 2/2 CFT]	1/2 LPR	1/3 HPI and 1/2 LPI	1/2 LPR	1/2 LPI and 1/2 CFT	1/2 LPR at 30 min when BWST depleted
10.0				1/3 HPI and	Same	1/2 LPI	Same as above	1/2 LPI and	Same		
13.5	1.0			1/3 HPI and 1/2 LPI and 2/2 CFT[2]	Same as above			1/2 LPI and 2/2 CFT	Same as above		

[1]Success criteria for containment heat removal are nearly the same in all cases. One of three (four for Midland and Arkansas) RBCU or one of two RBSS are sufficient.
[2]Oconee RSSMAP also considered the following criteria at a later stage:
1/3 HPI and 1/2 LPI and 2/2 CFT for 4- to 10-inch breaks; 1/2 LPI and 2/2 CFT for >10-inch breaks.
[3]The differantiation between small and very small LOCAs in BNL review was made for containment system responses, the success criteria are the same for both ranges in the BNL review.

Table 2.10 Comparison of Very Small LOCA and SGTR Success Criteria for the Long Term

LOCA Initiator	Success Criteria
Very small-break LOCA (<1.5 inch)	1/3 HPR or [RCS cooldown and 1/2 LPR] if BWST suffice for 12 hours.
Steam Generator Tube Rupture (~400 gpm)	[RCS cooldown and affected steam generator isolation and 1/3 HPR] or Replenishing BWST (success of HPI is implicit) or [RCS cooldown and 1/2 DHRS]

Table 2.11 Major Support Systems

	Oconee-3	ANO-1	MIDLAND-2
Ac power system	• 3 load divisions – 2 Keowee hydroelectric generators – 100-kV transmission system from the Lee steam station • Several bus interties	• 2 load divisions • 2 4160-V emergency diesel generators • Several bus interties	• 2 load divisions • 2 4160-V emergency diesel generators • Several bus interties
Dc power system	• 2 load divisions • 2 125-V batteries Backup from Unit 1 • Bus interties	• 2 load divisions • 2 125-V batteries • Limited bus interties	• 2 load divisions • 2 125-V batteries • Bus interties
Engineered safeguards actuation system	• Actuates high pressure system, low pressure system, reactor building cooling system, reactor building spray system, and service water system standby pump • 2 out of 3 logic actuates upon 3-psig/10-psig containment pressure or 1550-psig/550-psig RCS pressure	• Actuates high pressure system, low pressure system, reactor building cooling system, and several support system components • 2 out of 3 logic actuates upon 4-psig/30-psig containment pressure or 1500 psig RCS pressure	• Actuates high pressure system, low pressure system, reactor building cooling system, and several support system components • 2 out of 3 logic actuates upon 4-psig/30-psig containment pressure or 1500-psig RCS pressure.
Service water system	• 2 pumps • Provides required support system cooling for high pressure injection system, low pressure decay heat coolers, emergency feedwater system, reactor building cooling system, HVAC room cooling, component cooling system	• 3 pumps/2 pump trains • Provides required support system cooling for high pressure system, low pressure system, spray system, reactor building cooling system, HVAC room cooling, diesel generator cooling	• 5 pumps/2 trains (shared by units 1 and 2) • Provides required support system for component cooling system, reactor building cooling system, HVAC room cooling, diesel generator cooling, AFWS
Heating, ventilation, air conditioning systems (HVAC)	• Required for low pressure injection and spray pump rooms*	• Required for high pressure, low pressure, spray pump rooms • Required for ac and dc switchgear rooms	• Required for high pressure, low pressure, spray pump rooms • Required for ac and dc switchgear rooms

*No longer a requirement according to oral information given by DPC in a meeting held on 6/14 (BNL and NRC were present).

Table 2.11 Continued

	Oconee-3	ANO-1	MIDLAND-2
Integrated control system	• Controls proper coordination between reactor, steam generators, main feedwater, and turbine during normal operation • Recent design upgrade has essentially eliminated ICS-caused failures of safety systems	• Controls proper coordination between reactor, steam generators, main feedwater, and turbine during normal operation • Recent design upgrade has essentially eliminated ICS-caused failures of safety systems	• Controls proper coordination between reactor, steam generators, main feedwater, and turbine during normal operation
Instrument air	• Several systems require instrument air for proper operation, e.g., MFW, EFW	• Several non-safety systems require instrument air for proper operation • Safety-related components fail safe upon loss of instrument air	• Several non-safety systems require instrument air for proper operation • Safety-related components fail safe upon loss of instrument air

Table 2.12 Oconee Frontline vs Support System Dependences

	Ac Power	125-V dc Power	Engineered Safeguards System	Service Water System (LPSW)	Compressed Air System	Component Cooling System	Condenser Circulating Water System	Recirculating Cooling Water System	Integrated Control System	Heating, Ventilation, and Air Conditioning Systems	Primary Pressure Control System
Reactor protection system	✓	✓									
High pressure injection/recirculation	✓	✓	✓	✓							
Low pressure injection/recirculation/decay heat removal	✓	✓	✓	✓	✓					✓	
Power conversion system	✓	✓			✓		✓	✓	✓		
Emergency feedwater system	✓	✓		✓	✓		✓				
Pilot-operated relief valve	✓	✓							✓		✓
Reactor building cooling system	✓		✓	✓							
Reactor building spray system	✓	✓	✓	✓						✓	
Standby shutdown facility											
Core flooding system											
High pressure injection system - makeup mode	✓	✓		✓	✓	✓					

Table 2.13 Oconee Support vs Support System Dependences

	Ac Power	125-V dc Power	Engineered Safeguards System	Service water System (LPSW)	Compressed Air System	Component Cooling System	Condenser Circulating Water System	Recirculating Cooling Water System	Integrated Control System	Heating, Ventilation, and Air Conditioning Systems	Primary Pressure Control System	Human
Ac power	X	✓										
125-V dc power	✓	X										
Engineered safeguards system	✓	✓	X									✓
Service water system	✓	✓	✓	X	✓		✓	✓				✓
Compressed air system	✓			✓	X			✓				✓
Component cooling system	✓	✓		✓	✓	X						
Condenser circulating water system	✓						X					
Recirculating cooling water system	✓						✓	X				
Integrated control system	✓	✓			✓				X			
Heating, ventilation and air-conditioning system	✓			✓						X		
Primary pressure control system	✓	✓									X	

Table 2.14 Comparison of Transient Initiators and Their Grouping in the Arkansas IREP, Midland, and Oconee PRAs

Arkansas-IREP		Midland-PRA		Oconee-PRA	
Transient Initiator	EPRI-NP-2230 Grouping	Transient Initiator	EPRI-NP-2230 Grouping	Transient Initiator	EPRI-NP-2230 Grouping*
Turbine trip	1,2,3,6,10,14 15,17(50%),23 33,34,37,38,39	Reactor trip	1,2,3,8,11,12 14,15,17,21,22 23,28,36,37, 38,39,40	Turbine trip (T1)	1,2,3,4,5,6 7,8,10,11,12 13,14,26,27, 28,29,33,34, 36,37,38,39
		Turbine trip	18,33,34		
Total interruption of the PCS (Loss of main FW)	16,17(50%),18, 20,21,22,24, 25,29,30	Total loss of FW	16,24,25,27, 30	Loss of main FW (T2)	16,24
				Partial loss of FW (T3)	15,21,22,23
				Loss of condenser vacuum (T4)	25,30
		MUPS malfunction	---	Spurious ESF signal (T8)	9
		Excessive FW flow	---	Excessive FW flow (T7)	19,20
		ICS malfunction	---	Loss of ICS bus KI (T11)	---
				Low PZR pressure signal (T13)	---
Loss of service water		Loss of service water	---	Loss of service water (T12)	32
		Loss of component cooling	---		
Loss of offsite power		Loss of offsite power		Loss of offsite power (T5)	35
Loss of ac power bus A3 Loss of ac power bus B5 Loss of dc power bus D01 Loss of dc power bus D02				Loss of ac power bus 3TC (T14)	
				Loss of air (T6)	
		Steam-line break	---	Steam-line break (T9)	---
				FW-line break (T10)	---

*OPRA does not provide this grouping. They were generated from Table 5.8 by the BNL review.

3. ACCIDENT SEQUENCE DEFINITION

The objective of this section is to provide a discussion and major conclusions of the review on the following topics: 1) the OPRA[1] accident sequence definition and the qualitative description of the functional event trees (Section 3.1), 2) the system fault trees that were used in the OPRA (Section 3.2), and 3) the various aspects of human performance analysis that entered into the risk assessment (Section 3.3).

3.1 Functional Event Trees

Subsections 3.1.1 through 3.1.3 briefly present the general methodology, the functional event tree development, and the treatment of dependences used in the OPRA,[1] respectively; an overview of the BNL comments, where applicable, isalso presented. Subsections 3.1.4 through 3.1.6 present detailed discussions of the BNL qualitative review of the transients, LOCAs and steam generator tube rupture (SGTR), and anticipated transients without scram (ATWS), respectively. The analysis of the interfacing LOCAs for Oconee-3 does not require the use of event trees because it was assumed that the occurrence of the event results in core damage; therefore, in this report, the interfacing LOCAs are discussed in Section 5.

3.1.1 The General Methodology

An accident sequence is defined in the OPRA report, as "a sequence of events leading to a core damage state of interest, the resultant break of the barriers to the release of the radionuclides in the core, and the transport of those radionuclides after their release." In this BNL review only the development and definition of the portion of the accident sequences up to the onset of core damage are reviewed; this corresponds with Chapters 3 to 8 in the OPRA. This portion of an accident sequence is referred to as "core-melt sequence" in the OPRA, and as "accident sequence" or "core damage sequence" in this review.

Three steps were taken by the OPRA to identify as completely as possible the core damage sequences for Oconee-3: 1) a search for the initiating events of interest; 2) the formulation of a set of safety functions necessary to prevent core damage; and 3) a detailed analysis of the plant-system failures that preclude the success of these functions. To link these three steps to obtain the core damage sequences in a systematic manner, an event tree/fault tree methodology similar to that used in the Reactor Safety Study (RSS)[2] was employed. The OPRA used a variation of the RSS approach, also called the small event tree/large fault tree method. In this method a supporting logic, sometimes called functional fault tree or top-level fault tree, is developed for each top-event function in the functional event tree.

OPRA began the development of the event trees by constructing event sequence diagrams (ESDs) and translating the ESD actions into top events for event trees, with a different event tree constructed for each initiating event. This process became unmanageable because of the number and complexity of the event trees developed. Therefore, functional event trees were developed for transients, LOCAs, SGTR, and ATWS. The top events of these functional event trees represent the initiating events and the safety functions necessary to avert core damage. The initiating events are discussed in Section

2.2, and the definition of the safety functions is reviewed in Section 2.1 of this report.

Since, as stated above, the OPRA uses the small event tree/large fault tree approach to link the functions (or top events) of the functional event trees to the detailed analysis of the plant-system failures (hardware or human errors, and the different initiating events including their interaction with equipment unavailability), an intermediate step was necessary. This step was performed via the construction of the supporting logic in which the top events are the functional event tree top events and the inputs are top events from the system fault trees and human errors.

3.1.2 Functional Event Tree Development

The functional transient event tree starts with an initiator followed by the subcriticality function. The success or failure of this function has a dramatic effect on the ability to achieve the other safety functions considered in the functional event tree. Therefore, the sequences with failure of the subcriticality function are developed on separate event trees (ATWS event trees). The next function is the preservation of the RCS integrity. Again, the impact of success or failure of this function is large. Its failure is transferred to the small-LOCA functional event tree. The LOCA event tree represents breaks in the RCS integrity due to pipe breaks or to transient-induced LOCAs (i.e., PORV/SRV stuck open or RCP seal failure). The next functions are associated with removal of heat from the reactor core, transfer of heat from the RCS, and long-term core cooling. The end points of the functional event trees in OPRA can be one of the following.

a) Successful hot or cold shutdown, and cooldown.

b) Bin I core damage: Early failure of core cooling following a transient-induced LOCA or a small-LOCA initiator (within about two hours after initiation).

c) Bin II core damage: Late failure of core cooling following a transient-induced LOCA or a small-LOCA initiator (within about 12 hours after the onset of the LOCA).

d) Bin III core damage: Early failure of core cooling following a transient.

e) Bin IV core damage: Late failure of core cooling following a transient.

f) Bin V core damage: Early failure of core injection following a large LOCA.

g) Bin VI core damage: Late failure of core recirculation following a large LOCA.

h) Transfer to other sequences, which will then result in one of the above seven end points.

The core damage bins for SG tube rupture were designated IR and IIR.

A successful shutdown and cooldown is considered to be hot shutdown in most cases. Cold shutdown is considered as the most desirable end state for the SGTR sequences.

3.1.3 Treatment of Dependences

The treatment of dependences within the accident sequences is inherent to the methodology employed in the OPRA and followed by the BNL review. An accident sequence was modeled with the SETS[3] computer code by merging the initiating events, the top function in the event tree, its supporting logic, and the detailed system fault trees. The OPRA and the BNL review carefully used the same designators for the same basic events, and cross-reference of frontline system to support system was kept. Therefore, the minimal cut sets obtained for any sequences directly accounted for the system interdependences found in the analyses.

Human error dependences were reviewed on a sequence-by-sequence basis. To avoid elimination of potentially important sequences, high screening values for human errors were employed in the quantification of sequences (see Section 5); these screening values were later modified when the sequences were individually reviewed.

3.1.4 The Transient Functional Event Tree

The OPRA constructed only one functional event tree for all transient initiators. This section presents the review of this functional event tree, including its supporting logic. It is important to note that the review of the transient functional event tree is in complete agreement with the OPRA. Therefore, only a brief discussion is given in this section, because there is no need to repeat the detailed description given in the OPRA, Section 3.3.

The transient event tree is modeled using the functional event tree given in Figure 3.1, and the function-oriented top events are listed in Table 3.1. The supporting logic for the top events in the functional event tree is given in Figures 3.2 through 3.8. Again, it should be emphasized that very few differences exist between this review and the OPRA; so Figures 3.2 through 3.8 in this report are almost reproductions of Figures 3.5 through 3.11 in the OPRA. Some event names were modified because the OPRA uses what is called the modularized system fault trees and this review uses the detailed fault trees.

The first top event evaluates the question of successful scram (K). Where the scram is not successful, the sequences are transferred to the ATWS event trees (see Section 3.1.6). Otherwise, the evaluation of the function Q, loss of RCS integrity, follows.

Supporting logic for function Q (Figure 3.2) is developed to evaluate the different possibilities that could result in a stuck-open pressurizer PORV or SRV (gate Q02), or that could lead to a failure of the RCP seals (gate HPRCPF, transferred from the HPI system fault trees). The analysis of the progression of the transient accident sequences with failure of the RCS integrity, Q, is transferred to the event tree for small-break LOCAs (Figure 3.9).

Function B, failure to maintain heat removal from the core and RCS via the steam generators, is the next function. The development of the supporting logic for this function is given in Figure 3.3. The failure of the RCS heat removal would happen if:

a. All feedwater flow is lost (gate B02),
b. RCS circulation is lost (Gates B03 and B04), and
c. large feedwater line breaks occur (initiating event T_{10}).

The success of function B indicates the achievement of hot shutdown, and failure indicates a demand for forced cooling by the HPI system to avoid core damage (feed and bleed). This mode of cooling requires the opening of the PORV or one SRV, event P, to provide a path for decay-heat-removal cooling and the success of the HPI system, event U_T. The failure of event P (Figure 3.4) is conservatively considered to lead to a core damage; this is very conservative, but its contribution to core damage is negligible and this assumption is used for both the OPRA and the BNL review. Failure of the HPI system in the feed-and-bleed mode of operation is represented by U_T (Figure 3.5) and includes the operator's failure to actuate the system, a procedural step in the Oconee Emergency Procedures.[4]

The remaining events, Y_T, L, W, and X_T, all relate to possible effects after HPI cooling is successful (feed-and-bleed mode of operation).

Event Y_T (Figure 3.6), failure to maintain RCS makeup supply, occurs if the reactor building spray system (RBSS) is actuated, depleting the inventory in the borated water storage tank (BWST). The occurrence of event Y_T affects the timing of events L, W, and X_T, because the RBSS will empty the BWST much more rapidly than the HPI system alone. Event Y_T will occur if the reactor building cooling system fails (gate BCTOP), or if breaks (in feedwater condensate or steam line) inside containment actuate the RBSS and the operators fail to stop it in 30 minutes (YT02).

Event L indicates failure to recover RCS heat removal in the steam generators before the BWST is emptied (about two hours for sequences including Y_T and about 12 hours for $\overline{Y}_T$). If RCS heat removal is successfully recovered ($\overline{L}$), the pressurizer relief valves must be reclosed to establish stable hot shutdown conditions. Failure of any of these valves to reclose, event W, results in a small-break LOCA and the sequences are transferred to the small-LOCA event tree. Finally, if RCS heat removal is not recovered before the BWST inventory is depleted (L), the failure to establish a long-term mode of HPI cooling, event X_T, will result in core damage.

In conclusion, BNL's review of the OPRA functional event tree for transients found that it correctly represents a detailed qualitative model of the plant response.

3.1.5 The LOCAs Functional Event Trees

The OPRA constructed the following functional event trees for LOCAs.

- Small-break LOCA event tree.

- Large-break LOCA event tree.

- Steam Generator Tube Rupture (SGTR) event tree.

A supporting logic for the top events in the funtional event trees was also developed, when necessary, for each of the above event trees.

This review basically agrees with the event trees and supporting logic used in the OPRA except that the OPRA treats the entire spectrum of breaks up to four inches in diameter as small-break LOCAs. The OPRA does acknowledge that for breaks smaller than about 1.5 inches in diameter [called very small LOCAs (VS) in this review] with no steam-generator cooling, the RCS will not depressurize and the HPI automatic initiation would not occur. However, the OPRA claims that calculations indicate that in these cases the RCS will tend to pressurize to the point at which the pressurizer PORV and/or SRVs will cycle and the containment pressure will increase to the set point (3 psig) of automatic initiation of HPI on high containment pressure in less than one hour (the time at which the core uncovery would start). This review did not benefit from these calculations, and on the basis of other analyses it assumes that for very small breaks the manual actuation of HPI will be necessary when steam-generator cooling is unavailable. Therefore, two identical functional event trees were constructed for small and very small LOCAs. Note that in the quantification of the sequences (discussed in Section 5), this difference was found to have no effect on core damage frequency, and because of this only the event tree for small LOCAs will be briefly discussed.

3.1.5.1 Small-LOCA Event Tree

The first question in the small-LOCA event tree, Figure 3.9, refers to the subcriticality function (K). If this function fails, a small-LOCA ATWS occurs; this sequence was not further characterized in the ATWS event trees because its contribution to core damage is very small by comparison with other ATWS.

Following a successful reactor trip ($\overline{K}$), the availability of the HPIS (U_S) is asked. If the HPI fails, core damage occurs. Note that at this point (success of reactor trip), the transient sequences with loss of RCS integrity (TQ) are transferred in.

If HPIS is successful ($\overline{U}_S$), the function "failure to maintain RCS makeup supply" (Y_S) is analyzed. If this function fails, either because the RBCS fails or because the operator fails to terminate the RBSS, the inventory in the BWST will be depleted in about two hours and the operators must start the high pressure recirculation (function X_S with the occurrence of event Y_S; gate XS02 in Figure 3.10). Failure of this function (X_S) results in core damage. Note that, for the very small LOCAs in this review, the RBSS is not automatically initiated unless the RBCS also fails.

If function Y_S is successful, the inventory in the BWST will be depleted in about 12 hours; at this time, a failure of high pressure recirculation and low pressure recirculation, if feedwater to steam generators was available, (function X_S, gate XS03 in Figure 3.10) results in core damage.

3.1.5.2 Large-LOCA Event Tree

In the large-LOCA event tree (Figure 3.11), the first question is the availability of injection (U_A), and its failure leads to core damage. With successful injection, the failure of the low pressure recirculation (X_A) also results in core damage.

BNL and the OPRA are in complete agreement on the large-LOCA event tree.

3.1.5.3 SGTR Event Tree

The SGTR is treated in OPRA as a special case of a very small LOCA; Figure 3.12 shows the SGTR event tree. If all feedwater pumps are lost, the pressure in the RCS may not decrease to the ES actuation set point and, therefore, manual HPI actuation is assumed to be required in both the OPRA and the BNL review.

The requirements for long-term cooling of the SGTR sequences are different from those of other very small LOCAs, because cold shutdown is the most desirable mode of long-term stable condition. However, as discussed in Section 2, long-term cooling at hot conditions is also a stable end state for SGTR sequences.

The special supporting logic for the long-term cooling in the SGTR case was also accepted with no changes by BNL, and for completeness purposes it is given here as Figures 3.13 through 3.16; these figures are reproduced from Figures 3.19 through 3.22 in the OPRA.

3.1.6 ATWS Event Trees

As discussed in Appendix E of the OPRA, and in Subsection 2.2.1.3 of this report, all transient initiators were grouped in four classes for analysis of the ATWS accident sequences, i.e., turbine trip, loss of offsite power, loss of condenser vacuum, and loss of main feedwater. Thus, the OPRA presents four functional event trees for the ATWS analysis (Figures 3.17 through 3.20).

The ATWS sequences for B&W reactors have been studied in depth.[5-9] Several reports present different peak pressures for the case in which the reactor protection system fails to scram the reactor following a transient. The results in the report BAW-1610,[5] where it is shown that essentially the RCS components would survive a peak pressure smaller than 3900 psig, were used by the OPRA and also by this review as a basis for the analysis of ATWS sequences. For peak pressures higher than 3900 psig, the OPRA assumes that deformation in the RCS valves would increase unavailability of the systems used for injection of borated water from the BWST; this review agrees with this assumption.

Before the discussion of the ATWS event trees, it is important to provide the assumptions used in the OPRA and in several other PRA studies. The most important assumptions are described below according to the logical order of appearance in the OPRA ATWS event trees reproduced here in Figures 3.17 through 3.20 (Figures E-1 through E-4 in the OPRA).

1. Feedwater to Steam Generators - In the Crystal River PRA,[10] it is stated that if feedwater (MFW or EFW) is available to the steam generators (SG) the core can be maintained in a stable condition without need for boration.

   In the Midland[11] and ANO[12] PRAs, it is stated that with feedwater available to the SG, boration may be needed in certain cases (not explained) in order to maintain a stable state.

   In the NRC analysis,[9] it is assumed that during the 50% of the time that the moderator temperature coefficient (MTC) is favorable, the availability of feedwater to the SG requires boration in about 10 minutes in order to maintain stable conditions.

   In the OPRA[1] it is assumed that for the fraction of time that the MTC is smaller than the 95% value (-1.04E-5 Δk/k-°F), the availability of feedwater to the SG also requires boration; however, the time constraint is not like that of the NRC analysis.[9]

2. Moderator Temperature Coefficient - In the Crystal River,[10] ANO,[12] and Midland[11] PRAs, the moderator temperature coefficient is not part of the analysis. In the NRC analysis, if the MTC is not favorable (i.e., Service Level C will be exceeded 50% of the time) core damage results.

   In the OPRA, if the MTC is larger than the 95% value, i.e., -1.04E-4 Δk/k-°F, a LOCA will result. In this case, since the peak pressure may exceed the 3900 psig threshold, deformation in the RCS valves may occur and the probability of failure to inject borated water is assumed to be 0.1 in the injection phase and 0.1 in the long-term phase.

3. Primary Pressure Relief - In the NRC[9] analysis, no consideration is given to the failure of the pressurizer PORV to open for pressure relief. In the Crystal River,[10] Midland,[11] and ANO[12] PRAs, the failure of the PORV to open is included but it makes no difference in the unavailability of the injection function if the SRVs open; note that it was assumed in the ANO analysis that the PORV block valve is always closed, and the Midland PRA, on the basis of communication with B&W,[11] assumes that for any peak pressure above 3200 psig the vessel head will lift and a LOCA will occur; no deformation of RCS valves will occur.

   In the OPRA, the failure of the PORV, or of any of the SRVs, to open is assumed to cause a peak pressure bigger than 3900 psig, and a LOCA will result. For the unavailability of injection or long-term cooling, the same values in item 2 above (0.1) are used.

On the basis of the above information, this review is in qualitative agreement with the ATWS event trees constructed in the OPRA (Figures 3.17 through 3.20), and only the analysis of the turbine trip ATWS (Figure 3.17) will be discussed here.

The first two top events in Figure 3.17 evaluate the question of successful main feedwater, M, or emergency feedwater, L. For the case where both of these systems are not successful the OPRA assumes that a LOCA would occur and the injection of borated water from the BWST, function I, is still possible, and if this function is successful the question of long-term cooling, function X, is asked.

For the case where either main feedwater, M, or emergency feedwater is successful, the next top level fault, moderator temperature coefficient (MTC) smaller than the 95% value, event C, is asked. If the MTC is larger than the 95% value, a LOCA would occur and the availability of injection of borated water, I, and long-term core-cooling, X, are evaluated. When the MTC is smaller than the 95% value, C, the pressure relief function, P_O, follows. If the PORV or any one of the SRVs do not open, a LOCA results, and the functions I and X follow. If the relief valves do open, $\overline{P}_O$, the closure of the relief valves, P_C, is evaluated. If the relief valves do not stick open, injection of borated water, I, will still be needed to shut down the reactor. Failure of this function or failure of long-term cooling will result in core damage.

For the case in which any of the relief valves is stuck open, a LOCA occurs and injection of borated water, I, and successful long-term cooling are needed to avoid core damage.

3.2 System Fault Trees

The system fault trees are given in Appendix A of the OPRA. The following system fault trees are analyzed in the OPRA:

Ac Power System (AC)
Compressed Air System (CAS)
Core Flooding System (CFS)
Dc Power System (DC)
Engineered Safeguards System (ESS)
Emergency Feedwater System (EFS)
Heating, Ventilating, and Air Conditioning System (HVACS)
High Pressure Injection, Makeup, and Reactor-Coolant-Pump Seal Cooling System (HPIS)
Integrated Control System (ICS)
Low Pressure Injection/Recirculation System (LPIS)
Low Pressure Service Water System (LPSW)
Power Conversion System (PCS)
Primary Pressure Control System (PPCS)
Reactor Building Cooling System (RBCS)
Reactor Building Spray System (RBSS)
Standby Shutdown Facility (SSF)

For most of the systems above, the OPRA constructed a detailed fault tree and a modularized fault tree (or reduced fault tree). This modularization consisted of grouping into a single event large pieces of the logic that are independent of all other events in all the fault trees (this step was done by the analysts, not with a computer code like SETS[3]). In this review, only the detailed fault trees were reviewed because BNL intended to use the SETS[3] computer code to evaluate the accident sequences, and therefore the modularized fault trees.

A thorough review of each system fault tree was performed by BNL based on the drawings and information given in the OPRA[1] and the Oconee FSAR.[13] It is important to note that almost no logic or instrumentation diagrams were available to the BNL reviewers and thus some aspects of the review were not independently verified, e.g., logic for actuation of pumps or for the integrated control system. Even though these parts were not reviewed, a comparison of the level of detail used for these portions in other PRA studies[10-12] indicates that this omission does not seem to affect the results of this review.

The review of the OPRA system fault trees resulted in very small changes to several fault trees. Since none of these changes had any effect on the most important minimal cut sets for the systems or on the accident sequences, these modifications are not discussed.

In conclusion, the BNL review found that the OPRA fault trees adequately represent the failure modes for each system (based on the information described above), and the level of resolution in the OPRA detailed fault trees (down to component level, if data are available) is consistent with state-of-the-art PRA practice. Two states are considered for each component in the fault trees: either the component operates as designed or it fails. The following items were not included in the analysis of the failure of a component (or systems):

a. External events (including earthquakes, fires, floods, and tornados),
b. Sabotage,
c. Operator errors of commission.

The review of the effects of external events is analyzed in Volume 2 of this report. Items (b) and (c) above are outside the scope of the PRA.

3.3 Human Performance Analysis

A human-reliability analysis was conducted as part of the OPRA[1] and its conclusions are presented in Chapter 6 and Appendix C of that document. Four categories of human errors were used. Their usual location in the logic trees is shown in Figure 3.21, and the four categories are defined as follows:

a) Unavailability errors (U): errors that occur before an initiating event; also called latent errors. These errors result in a system or component being unavailable and as such they are modeled at the subsystem or component level in the system fault trees.

b) Inadvertent actions (I): errors that occur when the operator unintentionally defeats the functions of a system during the course of an event. Typically, they are modeled at the subsystem or component level.

c) Operator inhibits (OI): errors that occur when an operator intentionally defeats the function of a system during the course of an event, because of misdiagnosis. These errors are typically modeled at the system level, i.e., at the top of the system fault tree.

d) Operator fails to (OF): errors that occur when an operator fails to perform a necessary action during the event. These errors are typically modeled at the system level. The OPRA also models the failure to recover a system's operation or to find alternative systems. Even though these errors also belong to the OF category, they are referred to as "recovery errors" and are normally modeled at the accident-sequence level to ensure that any factors specific to the sequence are appropriately considered.

The important human errors, with a description of the required action, the time available for the action (when appropriate), and the assessed probability, are given in Table 3.2. The BNL review is basically in agreement with the modeling approach and quantification of human errors used in the OPRA. Therefore, the human errors and assessed probabilities given in Table 3.3 are mostly taken directly from Table C.5 of the OPRA, Appendix C; a comparison of the human error probabilities used in the OPRA with those used in previous PRAs is given in Section 6.5 of the OPRA. Some additional recoveries were listed in Table 3.2 to reflect BNL assessment of the time available for recovery of the instrument air system. These additions, as will be seen in Section 5, have an important effect on the frequency of core damage calculated in this BNL review. It is important to note that these changes in recovery times were discussed with Duke Power Company, and an agreement was reached.

3.4 References

1. A Probabilistic Risk Assessment of Oconee-Unit 3, NSAC/60, June 1984.

2. Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, NUREG/75-014, October 1978.

3. Worrell, R. B. and Stack, D. W., A SETS User's Manual for the Fault Tree Analyst, NUREG/CR-0465, Nov. 1978.

4. Oconee Nuclear Station Emergency Procedures, EP/0/A/1800/14 - Loss of Steam Generator Feedwater.

5. Analysis of B&W NSS Response to ATWS Events, BAW-1610, June 1980.

6. McBride, A. F. et al., Babcock & Wilcox Anticipated Transients Without Scram Analysis, BAW-10099, Rev. 1, May 1977.

7. Anticipated Transients Without Scram for Light Water Reactors, NUREG-0460, Nov. 1981.

8. Collier, R. P. et al., Selected ATWS Audit Calculations for Three PWR Designs, Battelle Columbus Laboratories, Dec. 1982.

9. Recommendations of the ATWS Task Force, Enclosure D, NRC.

10. Garcia, A. A. et al., Crystal River-3 Safety Study, NUREG/CR-2515, Dec. 1981.

11. Midland Nuclear Plant - Probabilistic Risk Assessment, Consumers Power Company and PLG Inc., May 1984.

12. Kolb, G. J. et. al., Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit 1 Nuclear Power Plant, NUREG/CR-2787, June 1982.

13. Oconee Final Safety Analysis Report - Duke Power Company.

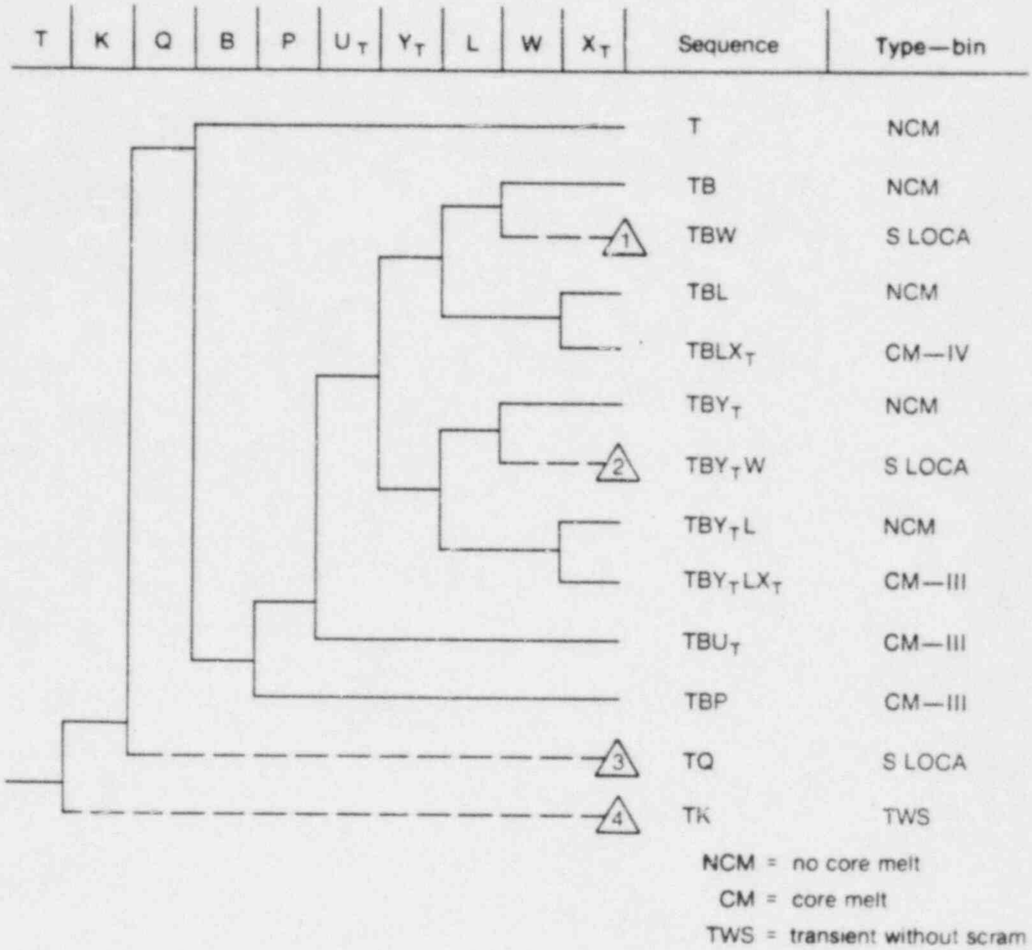


Figure 3.1 OPRA event tree for transient initiating events.

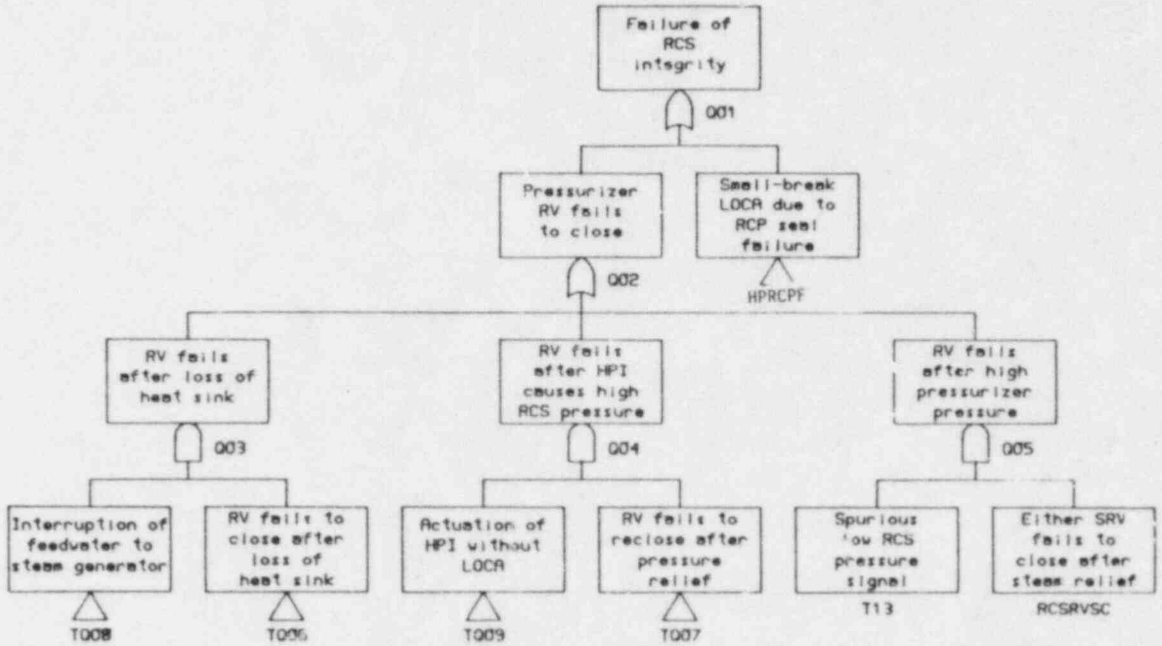


Figure 3.2 Supporting logic for transient event tree top event Q, failure of RCS integrity.

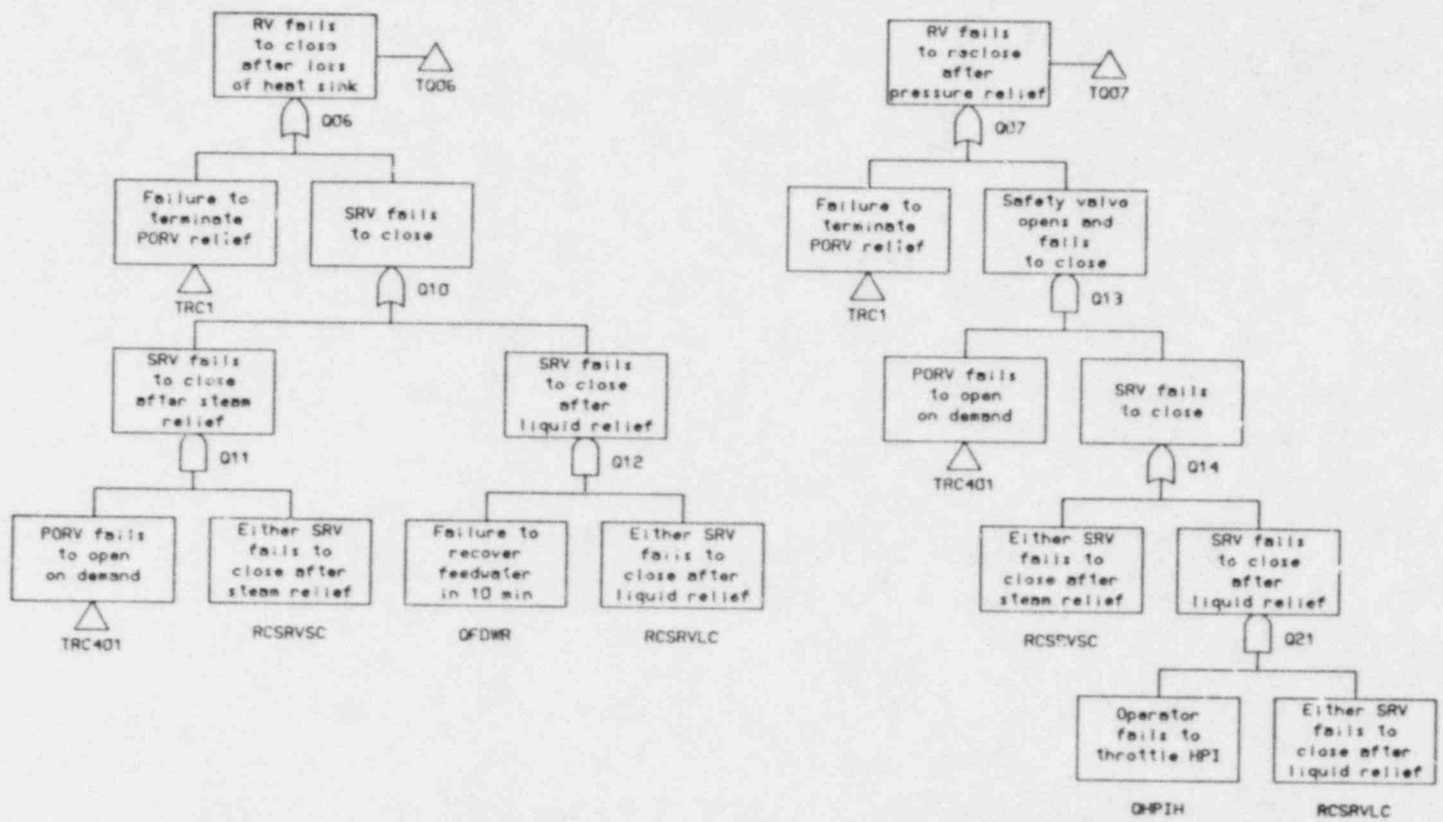
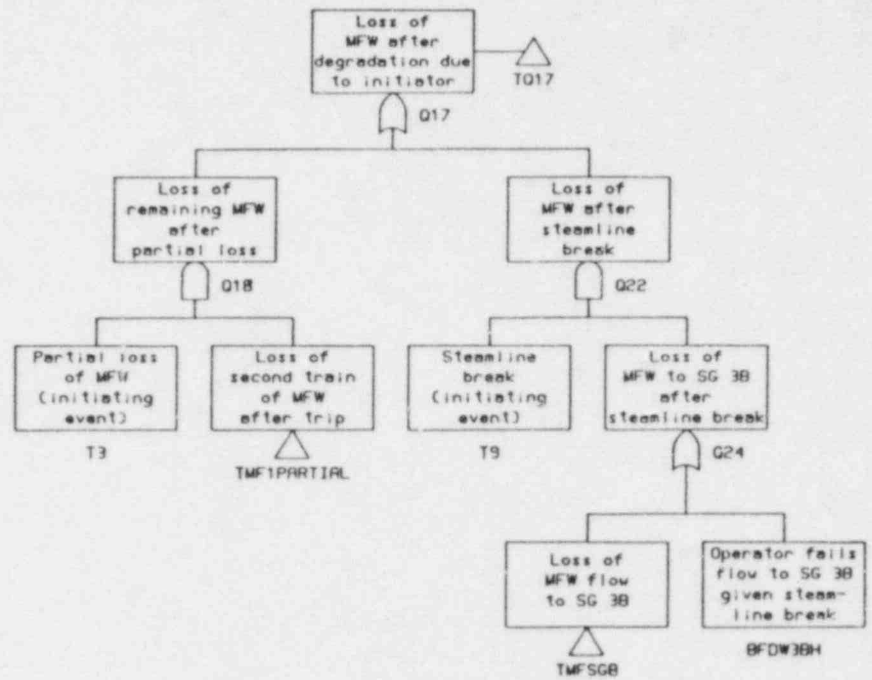


Figure 3.2 Continued



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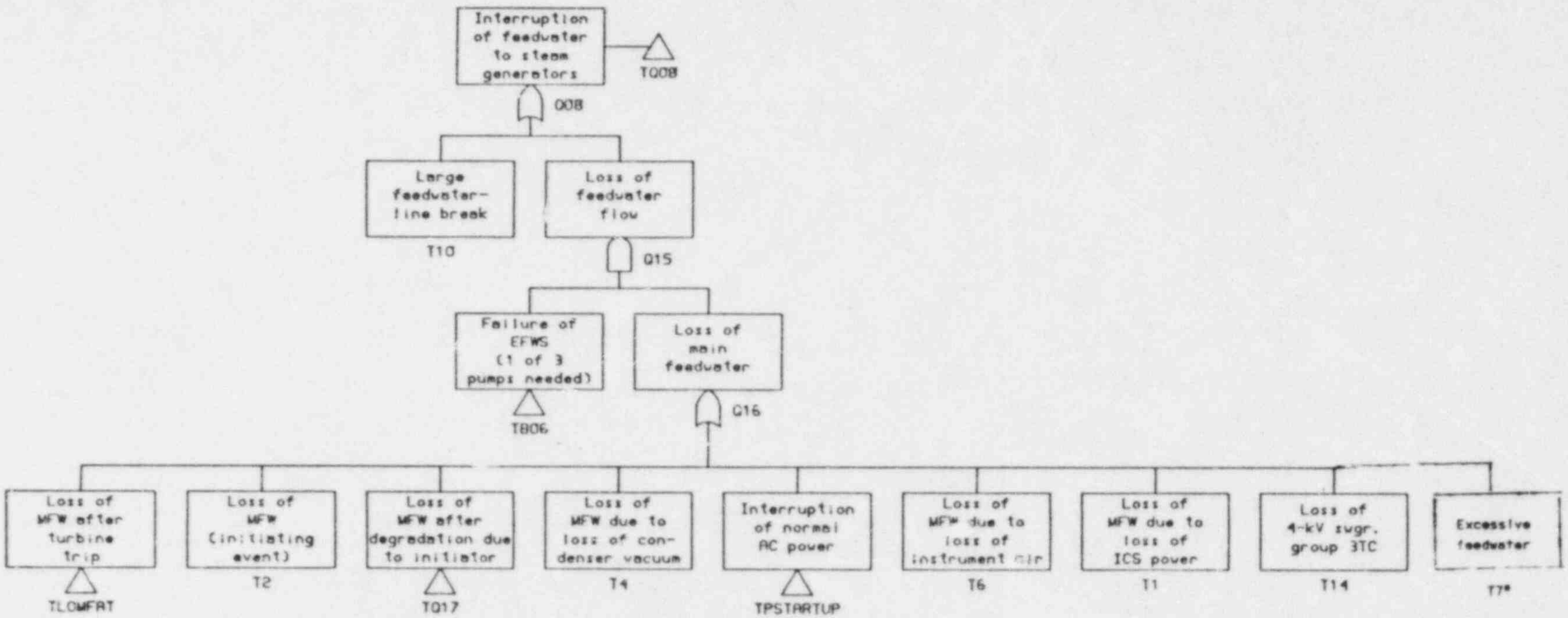
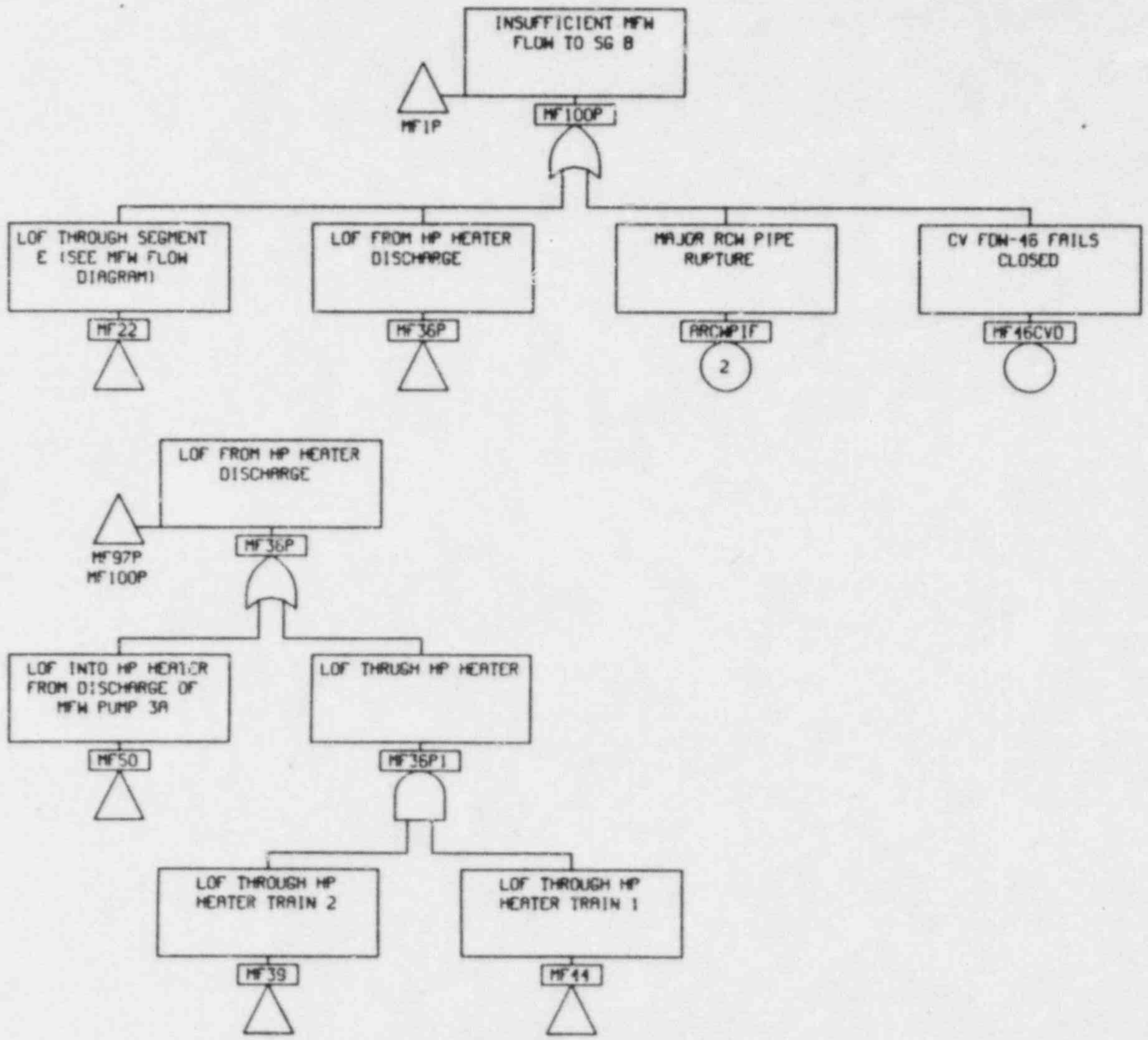


*With successful trip of the main feedwater pumps on high SG level ($\overline{MFWPRF}$)

Figure 3.2 Continued



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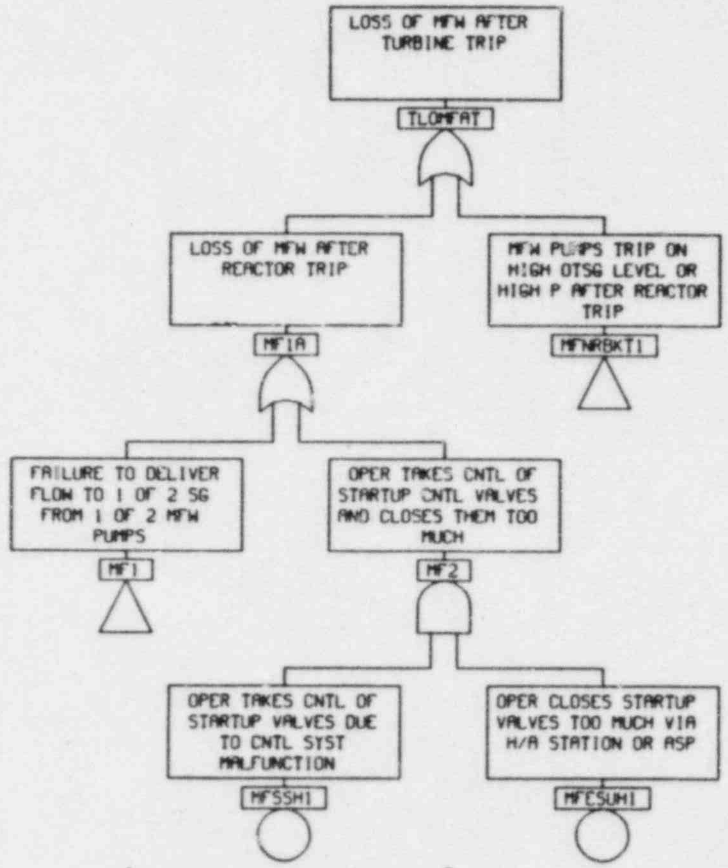

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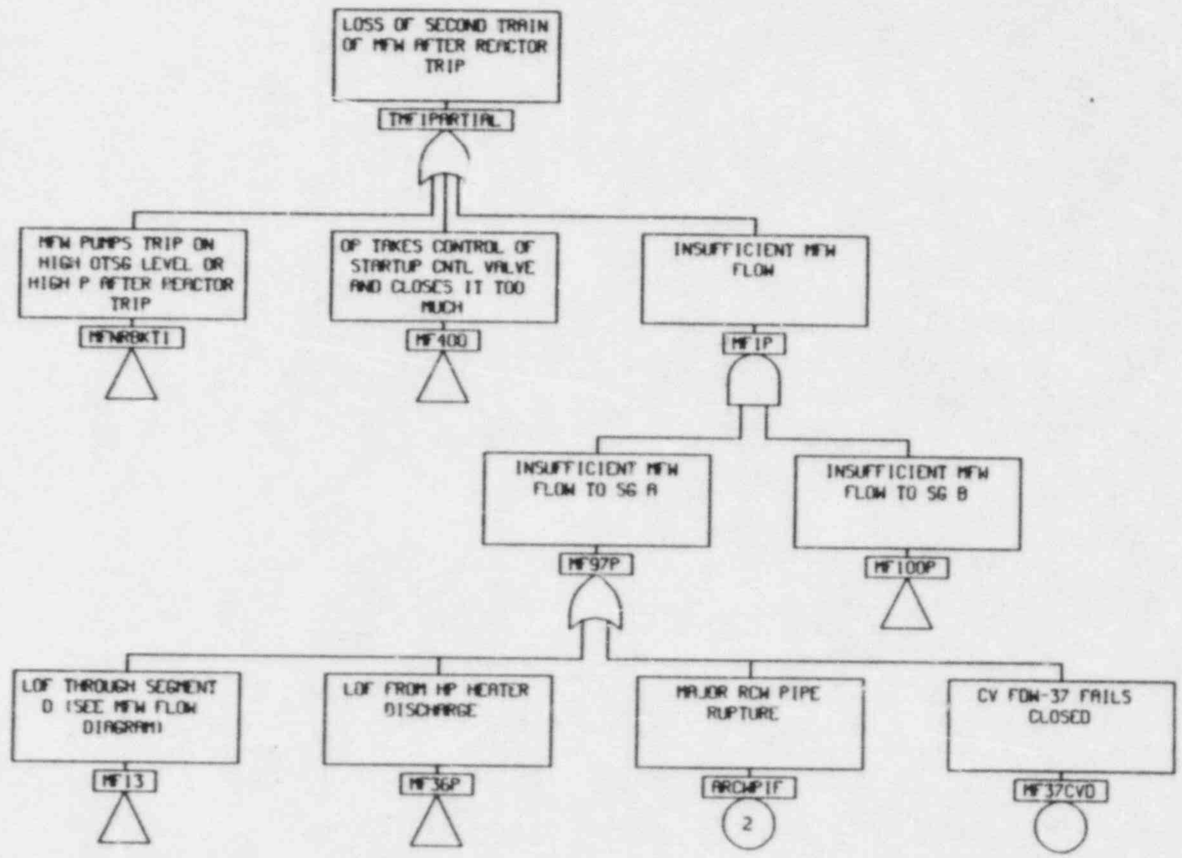

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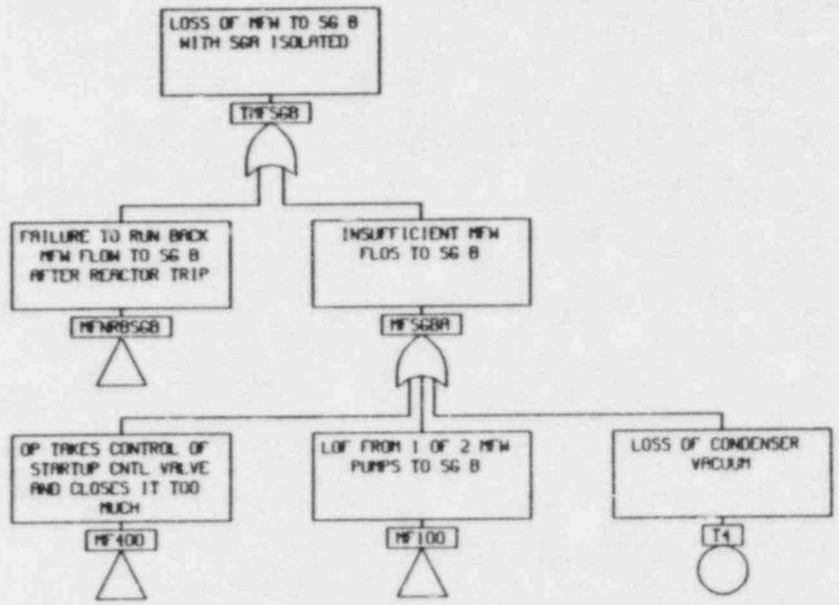

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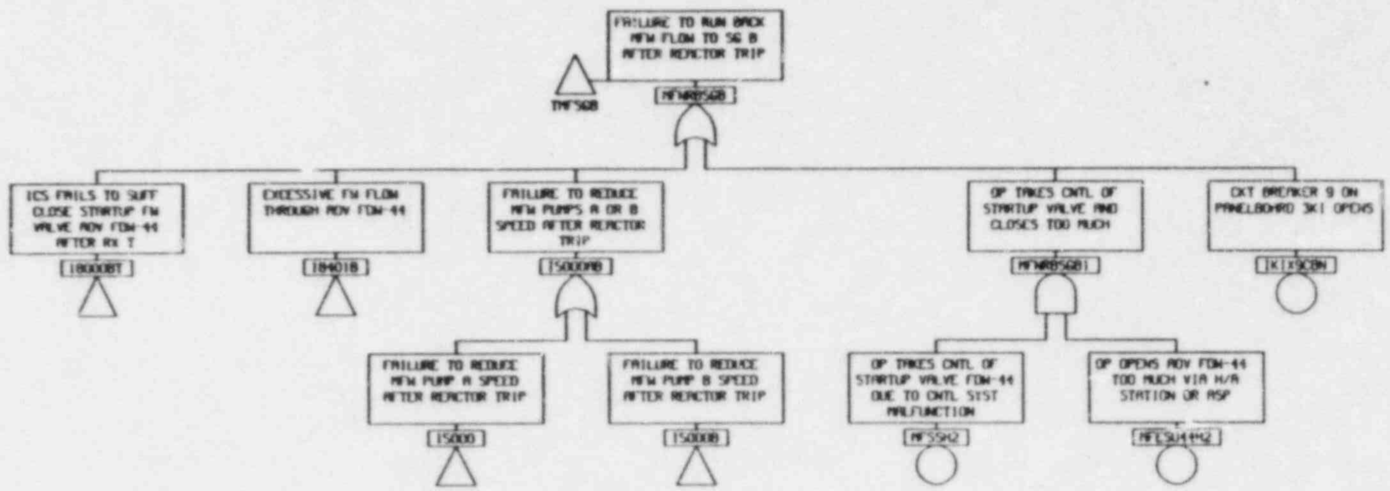

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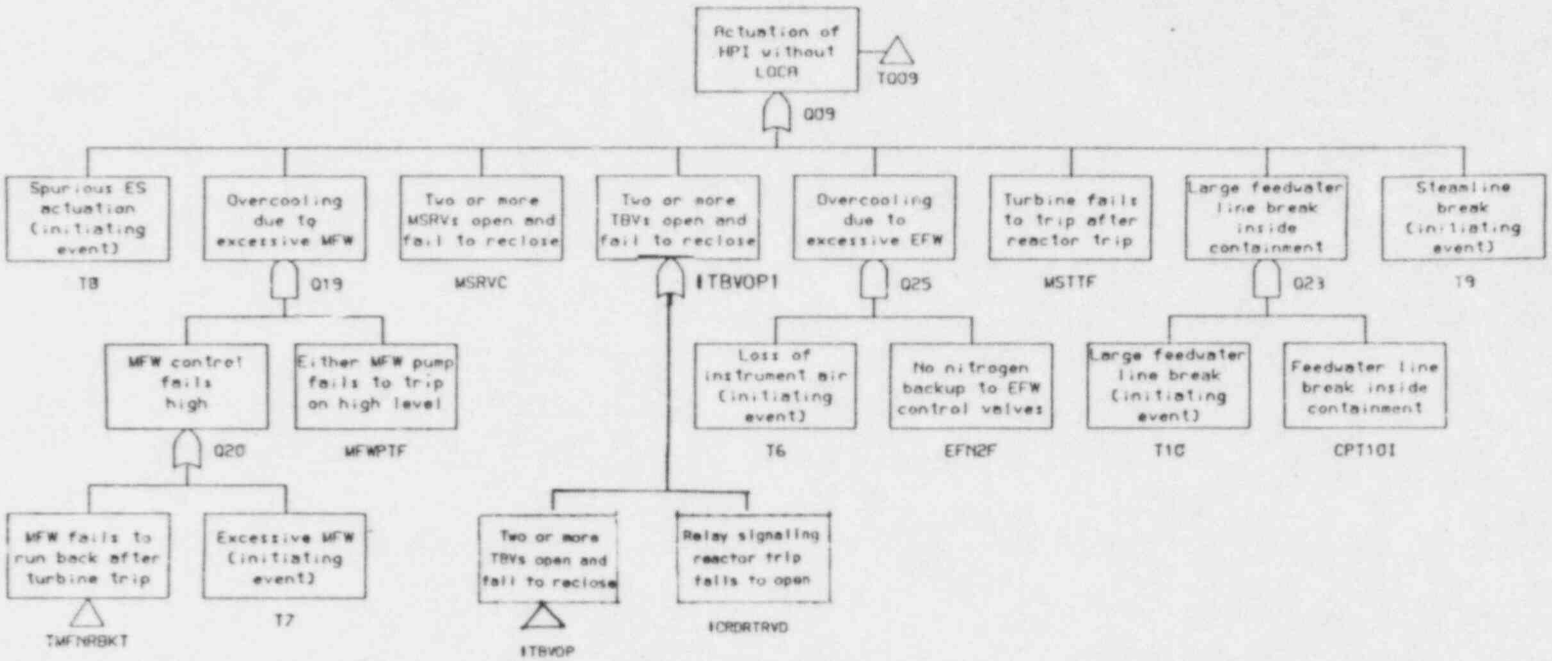

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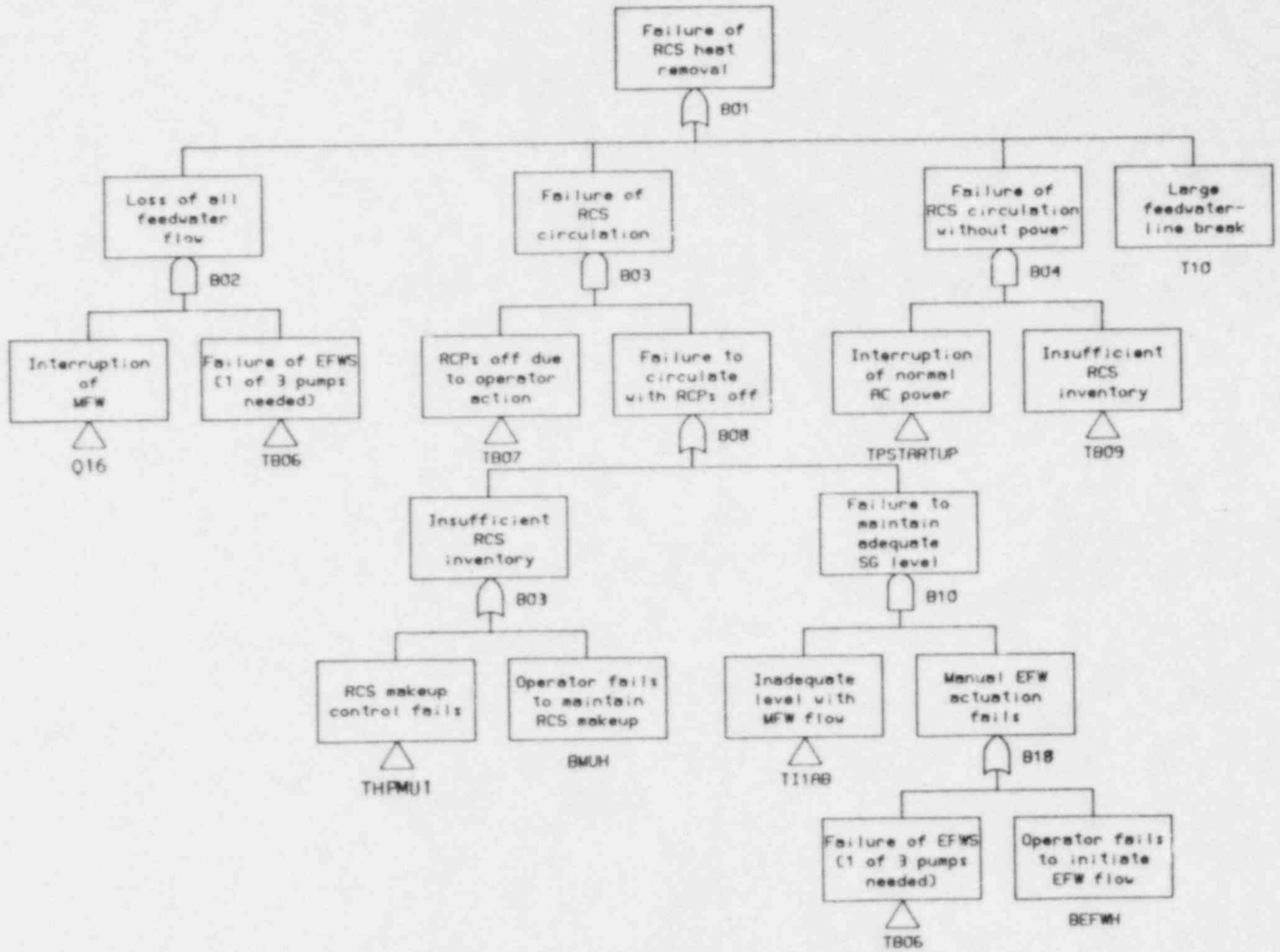


Figure 3.3 Supporting logic for transient event tree top event B, failure of RCS heat removal.

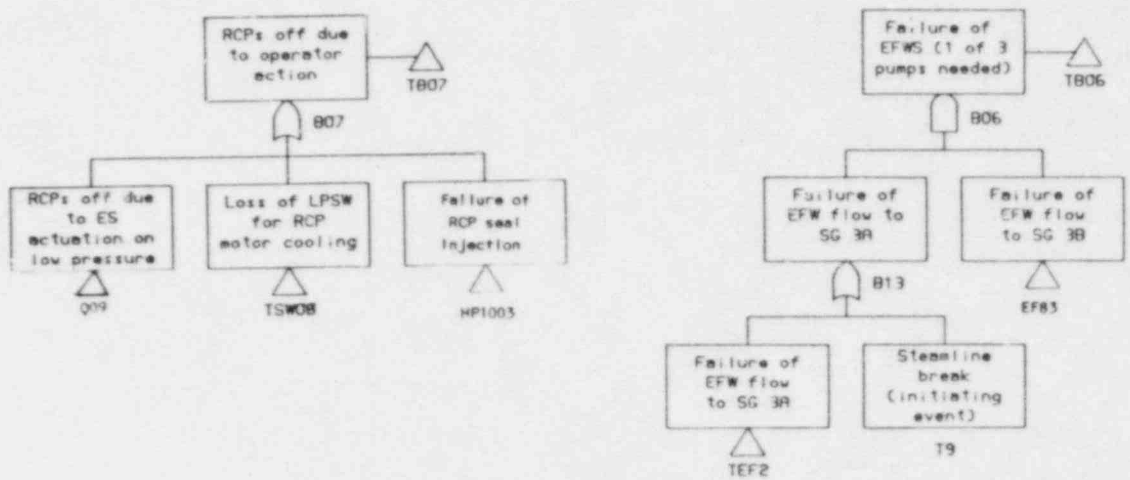


Figure 3.3 Continued

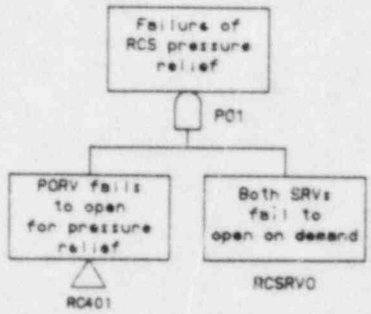


Figure 3.4 Supporting logic for transient event tree top event P, failure of RCS pressure relief.

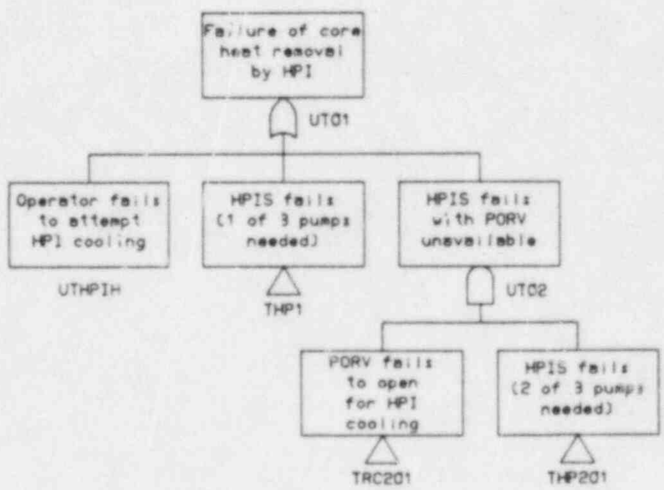


Figure 3.5 Supporting logic for transient event tree top event U_T, failure of core-heat removal by high pressure injection.

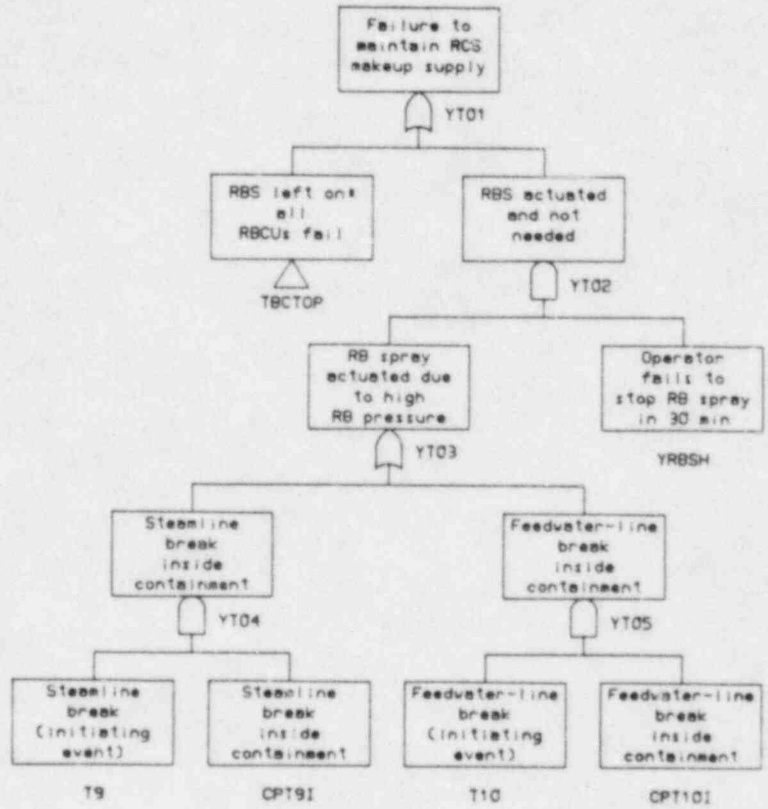


Figure 3.6 Supporting logic for transient event tree top event V_T, failure to maintain RCS makeup supply.

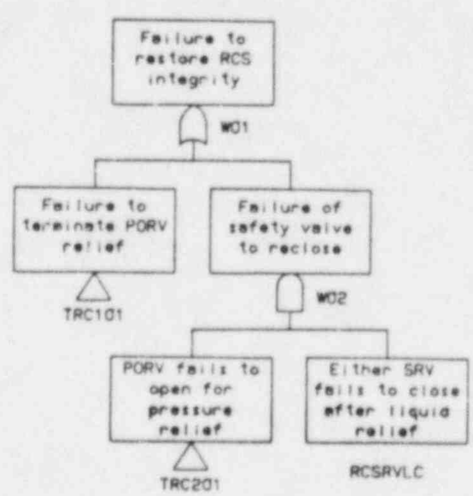


Figure 3.7 Supporting logic for transient event tree top event W, failure to restore RCS integrity.

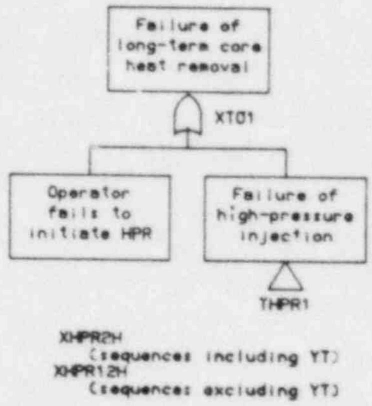


Figure 3.8 Supporting logic for transient event tree top event X_T, failure of long-term core-heat removal.

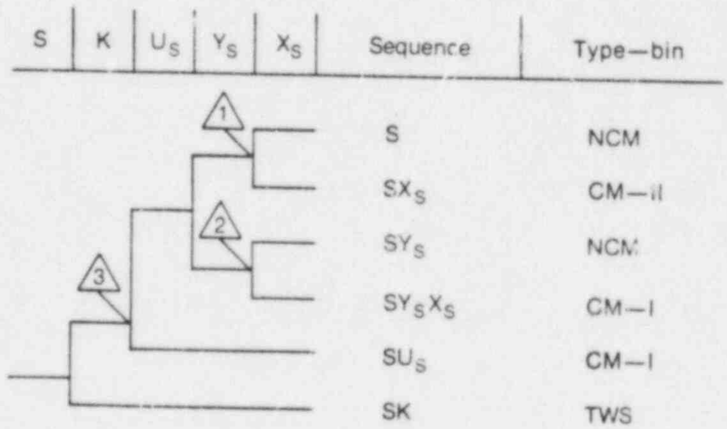


Figure 3.9 OPRA event tree for small-break LOCA events.

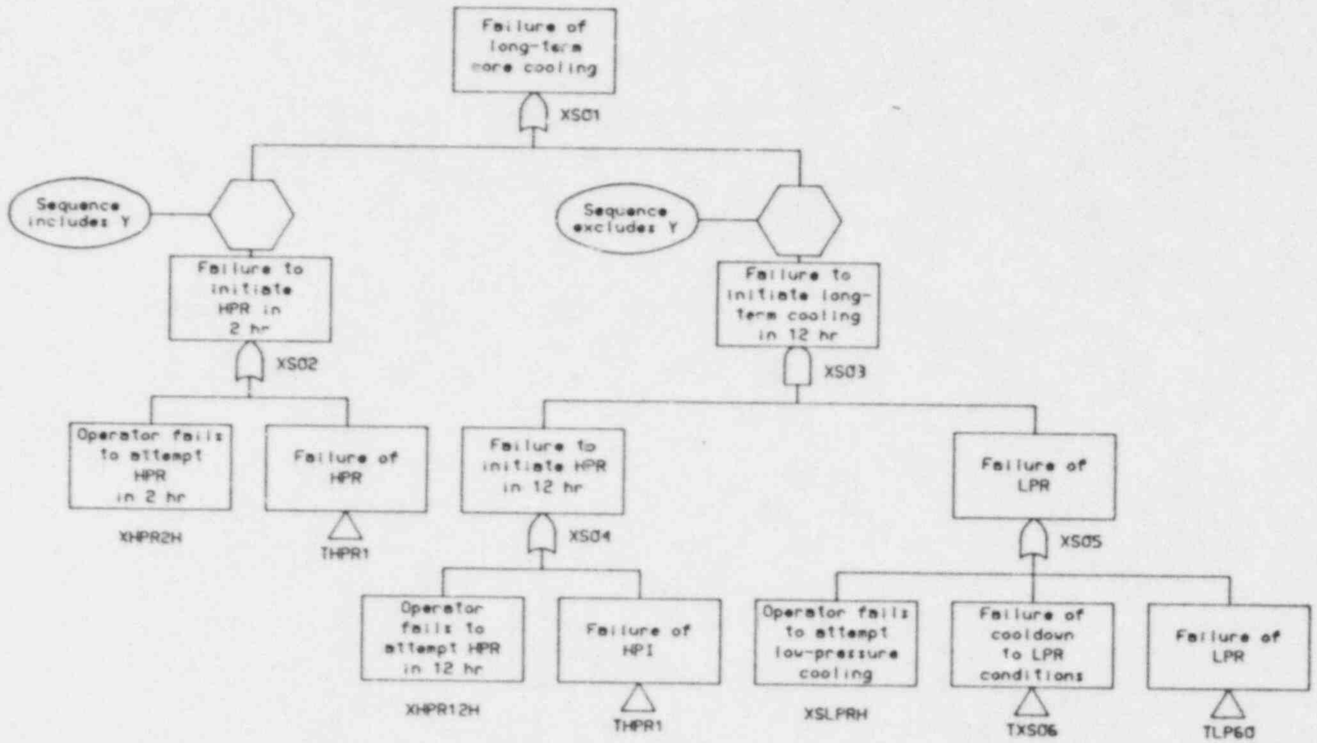
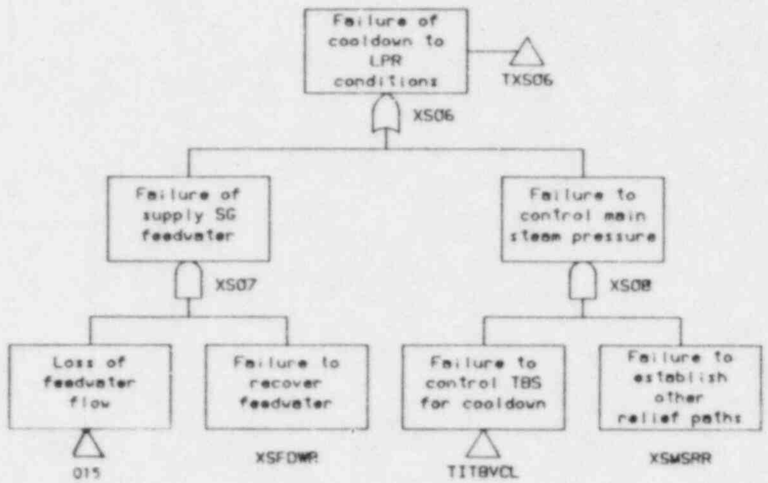


Figure 3.10 Supporting logic for small-break LOCA event tree top event X_S, failure of long-term cooling.



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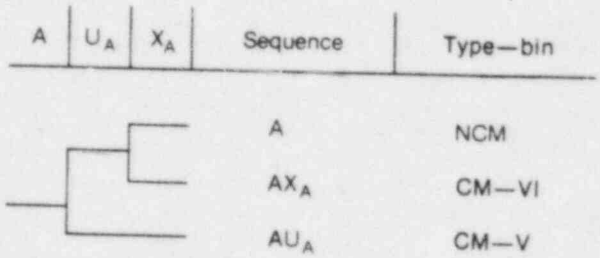


Figure 3.11 OPRA event tree for large-break LOCA events.

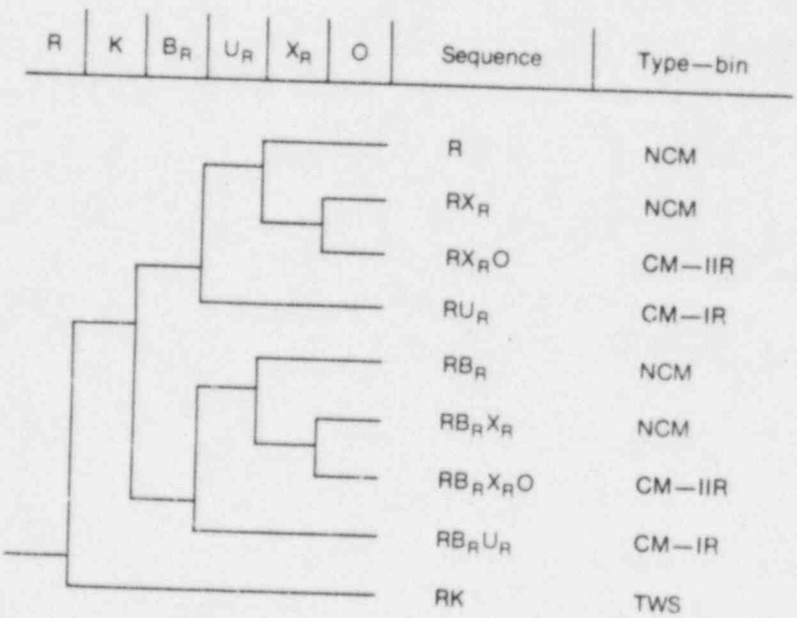


Figure 3.12 OPRA event tree for SGTR initiating events.

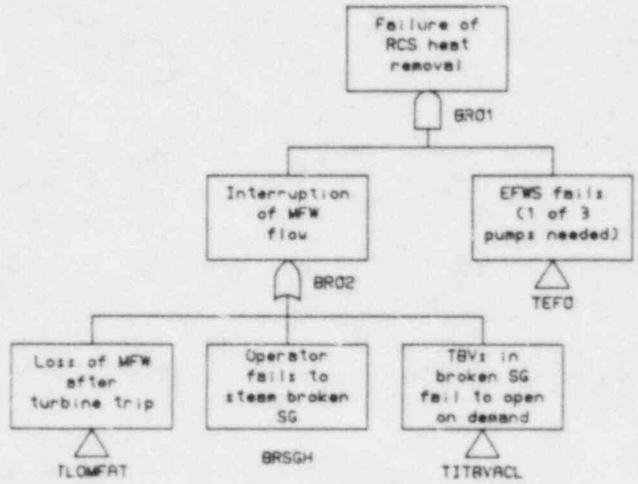


Figure 3.13 Supporting logic for SGTR event tree top event B_R, failure of RCS heat removal.

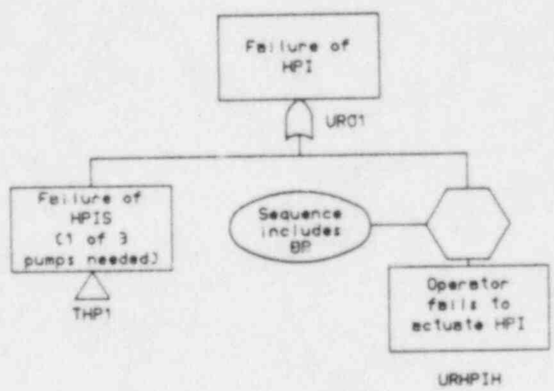


Figure 3.14 Supporting logic for SGIR event tree top event U_R, failure of RCS heat removal.

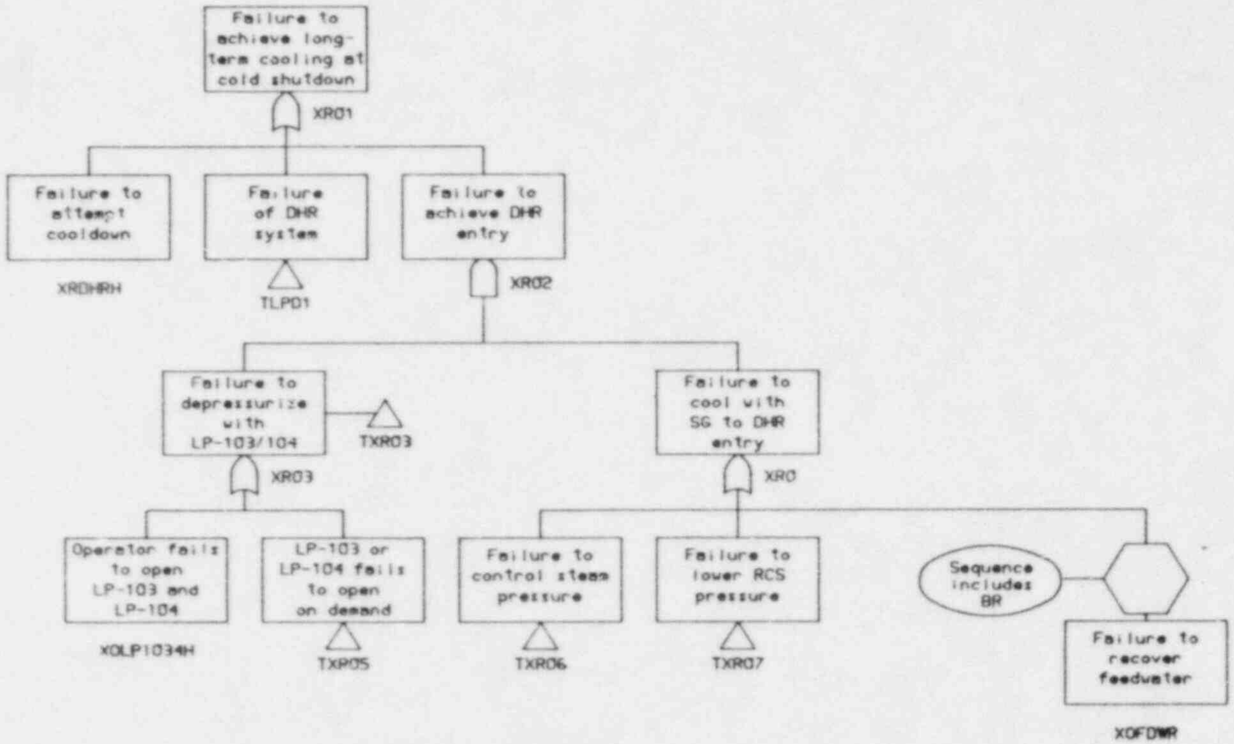


Figure 3.15 Supporting logic for SGTR event tree top event X_R, failure to achieve long-term cooling at cold shutdown.

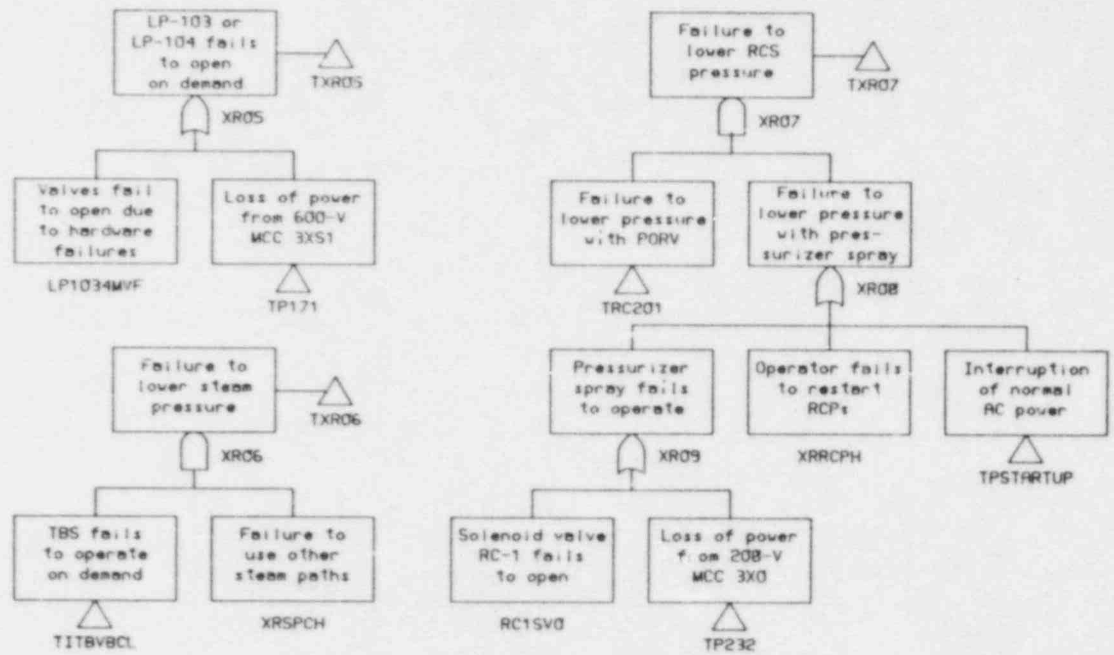


Figure 3.15 Continued

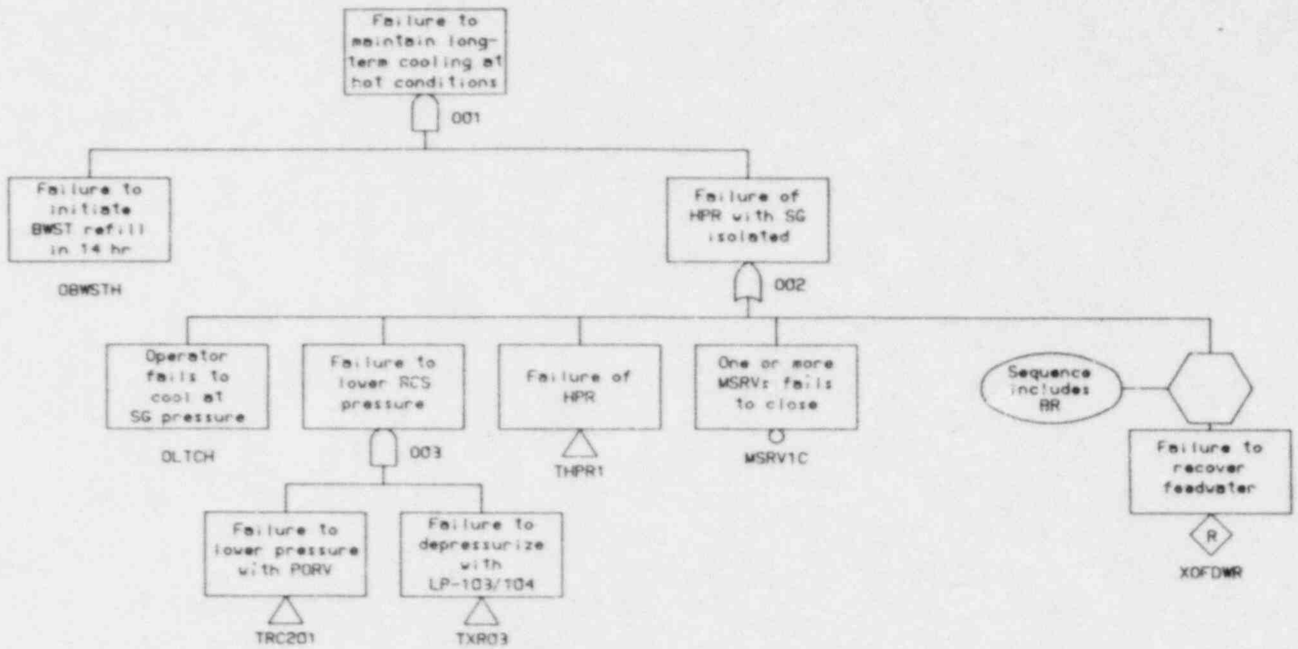
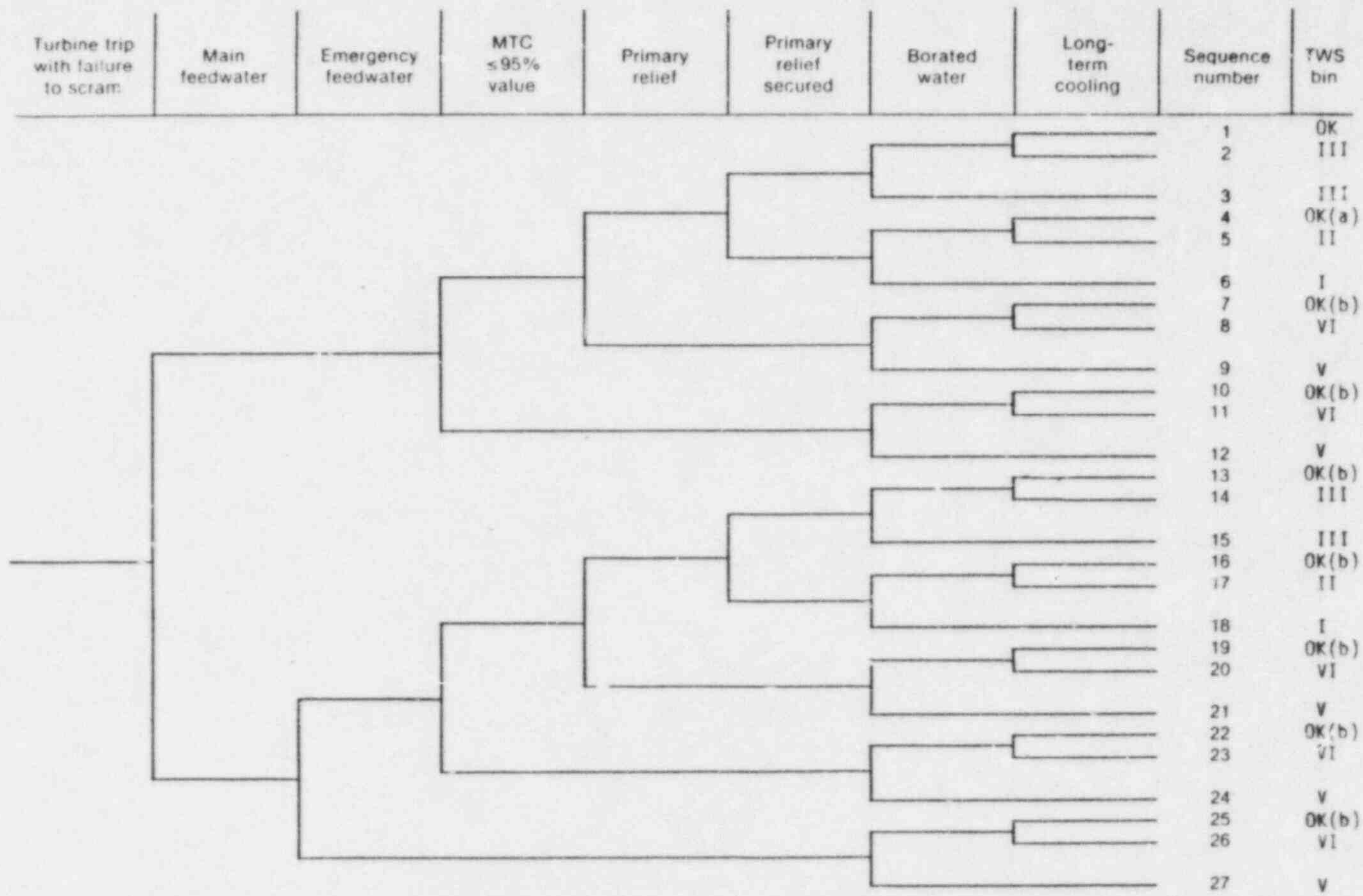
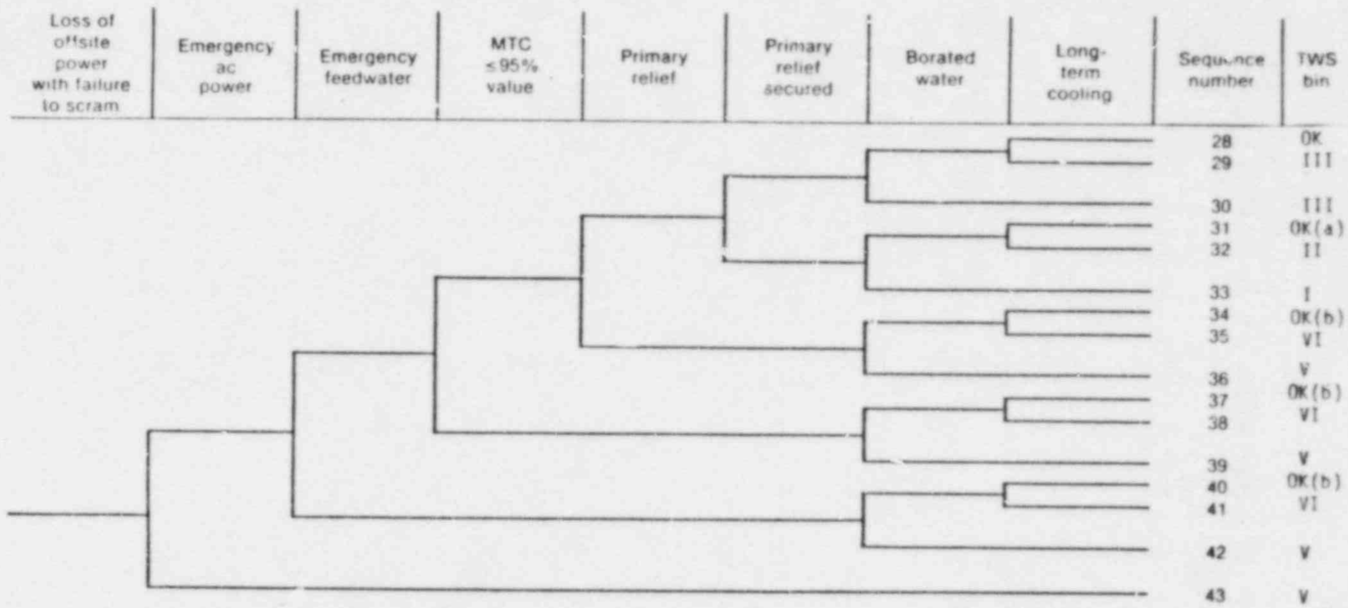


Figure 3.16 Supporting logic for SGTR event tree top event X_e, failure to maintain long-term cooling at hot conditions.



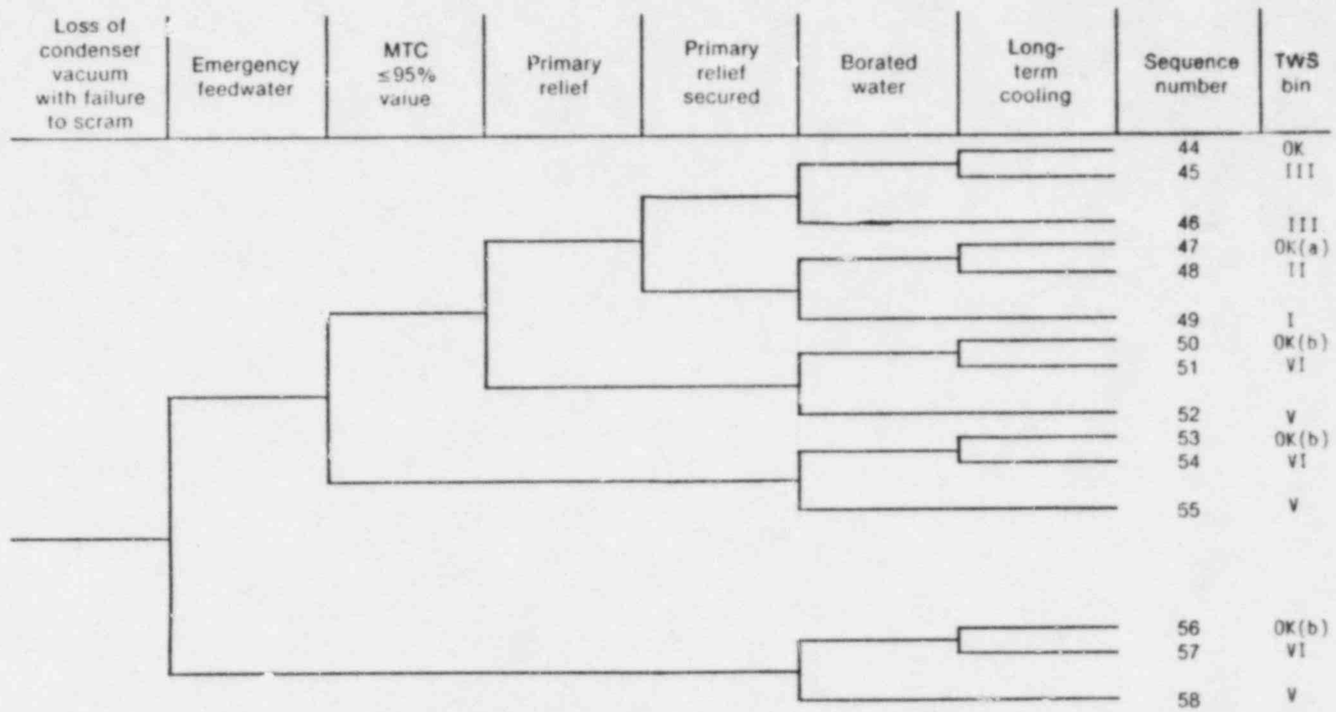
Note: OK = no core melt or LOCA
OK(a) = relief-valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure 3.17 OPRA scoping event tree for turbine trip with failure to scram.



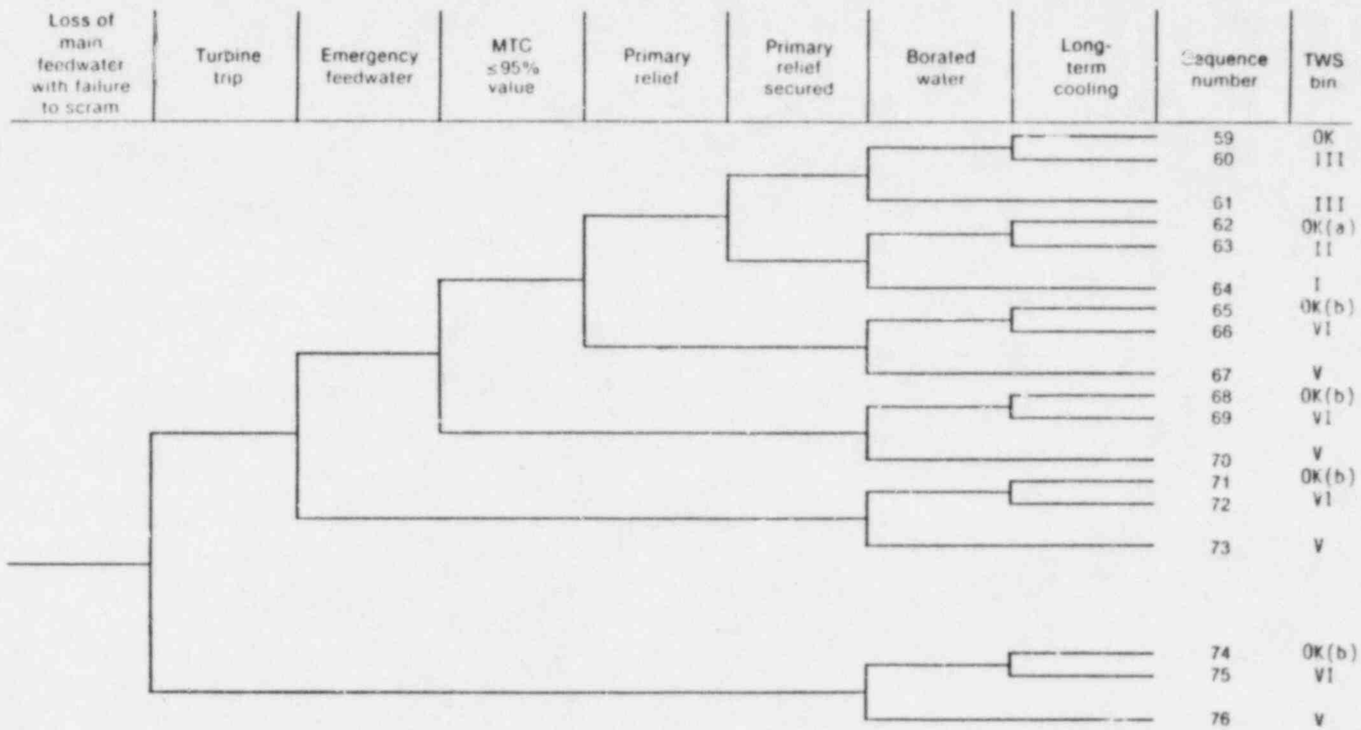
Note: OK = no core melt or LOCA
OK(a) = relief-valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure 3.18 OPRA scoping event tree for loss-of-offsite power with failure to scram.



Note: OK = no core melt or LOCA
OK(a) = relief-valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure 3.19 OPRA scoping event tree for loss of condenser vacuum with failure to scram.



Note: OK = no core melt or LOCA
OK(a) = relief valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure 3.20 OPRA scoping event tree for loss of main feedwater with failure to scram.

Table 3.1 Top Events in the Transient Event Trees

Event	Description
T	Occurrence of a transient initiating event
K	Failure of the RPS to trip the reactor
Q	Loss of RCS integrity
B	Failure of RCS heat removal via the steam generators
P	Failure to provide RCS pressure relief
U_T	Failure of core-heat removal by HPI cooling
Y_T	Failure to maintain RCS makeup supply
L	Failure to recover RCS heat removal
W	Failure to reestablish RCS integrity
X_T	Failure to maintain long-term core-heat removal

Table 3.2 Important Human Errors in the OPRA

Event	Description	Assessed Probability
YRBSH	Operator fails to terminate RB spray (given RB cooling available) to extend availability of suction for HPI during small-break LOCA	0.5
	Operator fails to initiate HPR after a small LOCA:	
XHPR2H	In 2 hr	0.003
XHPR12H	In 12 hr	0.0003
UTHPIH	Operator fails to attain or maintain HPI cooling in about 40 min after the loss of all feedwater	0.01
	Operator fails to recover feedwater in 30 min:	
REFDW1	One alternative available	0.5
REFDW2	Two alternatives available	0.3
REFDW3	Three alternatives available	0.1
	Operator fails to recover instrument air:	
REIA1	In 1 hr	0.5
REIA90	In 90 min	0.4
REIA2	In 2 hr	0.3
REIA6	In 6 hr	0.04
REIA12	In 12 hr	0.0024
TREFWSUC[a]	Failure in transfer of EFW suction to hotwell	0.15
CW157VV1H[a] CW391MV2H[a]	Operator fails to maintain suction supply to the steam-driven EFW pump by opening valve CW-391 and closing valve CW-157 (totally coupled)	0.1
CW391MVH[a]	Valve CW-391 not restored after maintenance	0.001
RESW12	Operators fail to recover LPSW from another unit before failure of the HPI pumps (in about 30 min after RCP trip)	0.014
RESW108	Operator fails to recover LPSW to the HPI pumps given valve LPSW-108 transfers closed (discharge path blocked)	0.11
RECCW73	Operator fails to recover LPSW to the HPI pumps given valve CCW-73 transfers closed (discharge path assumed to be blocked)	0.11
RESSFS1	Operator fails to provide RCP seal injection from the SSF within 30 min of losing seal cooling via HPI	0.1
LPTHROTTLE	Operator fails to throttle injection valves to prevent pump runout during LPR for a large-break LOCA	0.003
XALPRH	Operator fails to achieve LPR after a large-break LOCA	0.005
OBWSTH	Operator fails to refill BWST after an SGTR	0.5
RECC8	Operator fails to jack open AOV CC-8 after closure due to loss of instrument air	0.2

Table 3.2 Continued

Event	Description	Assessed Probability
REDHRSUC	Operator fails to locally open suction valves to permit motor operators	0.1
REEMOD12	Operator fails to restore ac power by manually closing an "E" breaker	0.1
	Operator fails to restore power to load-shed IA compressors (after loss of offsite power due to failure of feeders to the Oconee switchyard)	
REFEEDAIR1	In 1 hr	0.013
REFEEDAIR90	In 90 min	0.011
REFEEDAIR2H	In 2 hr	0.011
REFEEDAIR12H	In 12 hr	0.0012
REHPPPCS	Operator fails to allow standby HPI pump C to remain idle until restoration of pump cooling	0.05
RESSFW12H	Operator fails to initiate ASW from the SSF in 12 hr (human portion)	0.001
RESSFW30	Operator fails to initiate ASW from SSF in 30 min from loss of feedwater	0.1
	Operator fails to restore power to load-shed IA compressors (after loss of offsite power due to failure of the Oconee switchyard):	
RESUBAIR1	In 1 hr	0.034
RESUBAIR90	In 90 min	0.025
RESUBAIR2	In 2 hr	0.022
RESUBAIR12	In 12 hr	0.004
RESUMPMF	Operator fails to locate and isolate leakage from emergency sump via valves LWD-99 and LWD-103 before flooding HPI pump motors	0.1
SW71VVH[b] SW72VVH[b]	Valves supplying LPSW to LPI coolers left unavailable after maintenance	0.002
LPPSTOPH	Operator fails to turn off LPI pumps to prevent dead-heading during small-break LOCA	0.0008
LP28VVCH[b]	BWST suction valve LP-28 left closed after maintenance	0.000028
LWD99103H[b]	Valves LWD-99 and LWD-103 left open after maintenance	0.0006
BRSGH	Operator fails to steam the affected steam generator during an SGTR	0.01
HP2425MVH[b]	MOVs HP-24 and HP-25 (HPI ES suction valves) left unavailable	0.00005
LP15MVMH[b]	MOV LP-15 not restored after maintenance	0.0018
LP16MVMH[b]	MOV LP-16 not restored after maintenance	0.0018
SW3BPPSH[b]	Operator fails to start standby LPSW pump	0.008
SW7778CMH[b]	Manual valves LPSW-77 and -78 both left closed	0.003
SWC89VVH[b]	Manual valve CCW-89 inadvertently left closed	0.0008
SWEECCH[b]	Manual valves LPSW-513 and -518 inadvertently left overthrottled	0.0002

Table 3.2 Continued

Event	Description	Assessed Probability
XRDHRH	Operator fails to success[illegible]y initiate DHR after an SGTR	0.0003

[a]All these actions plus hardware failures are included in the TREFWSUC.
[b]Unavailability errors (U).

4. DATA ASSESSMENT

This section reviews the numerical values of the parameters necessary for the quantification of the accident sequences.

Subsection 4.1 includes the OPRA frequencies for the initiating events and the BNL reassessment; comparison with other PRAs is also given. The data base used in the OPRA for component failure rate and maintenance, with BNL comments and modifications (if appropriate), is presented in Subsections 4.2 and 4.3, respectively.

4.1 Frequencies of Initiating Events

4.1.1 The Quantification of Initiating Events in OPRA

The OPRA[1] considered 21 initiating events, as discussed in Section 2.2. Their frequency quantification was performed using two general approaches:

a. Use of generic nuclear power plant data to obtain a population prior and updating it with Oconee-specific experience. Most of the transients with scram were treated in this way.

b. Use of special studies such as a fault tree analysis, component failure data, or experience from other industries to evaluate the frequencies of plant-specific initiators, e.g., loss of air and loss of LPSW. In most of these cases, the basis for performing a special study was the belief that plant-specific design characteristics would provide a more realistic assessment than operating experience taken from other not always similar plants.

The quantification by case (a) approach sometimes lacks sufficient supporting information in the OPRA. The main missing link was the "grouping matrix" used in the reduction of the EPRI NP-2230[2] experience from 41 transients into the OPRA set of initiators.

The initiating-event frequencies used in OPRA are shown in Table 4.1, as are BNL values with differences and comments; further discussion is presented in the next subsections.

4.1.2 BNL Assessment of the Initiator Frequencies

4.1.2.1 Transient Initiators With Successful Scram

In this subsection, BNL discusses the transient initiators that were evaluated on the basis of EPRI NP-2230.[2] This report includes 36 PWR plants and a separate section on B&W plants in particular. It provides data on 41 different transient initiators for these plants.

The grouping of the 41 EPRI transient initiators into the corresponding OPRA event categories is not provided. BNL derived this grouping, and the results are given in Table 2.14, where a comparison with the grouping reported in Midland[3] and Arkansas[4] PRAs is also provided.

The following points can be made on the basis of this comparison:

1. Transients of MSIV closure (EPRI Nos. 17 and 18) are not considered in OPRA because this plant does not have MSIVs.

2. Manual trips by the operator (EPRI transient No. 40) are not considered in OPRA. It is considered as part of turbine trip in MPRA.

3. OPRA treats transients with PCS interruption more realistically by dividing them into four subgroups.

4. Midland PRA chose to treat the "low pressurizer pressure" transient (EPRI No. 6) as events leading to very small LOCAs. OPRA does not have this category. It is covered by the small-LOCA group for breaks ranging from 0.5 to 4 in. [see Section 4.1.2.3(a) for further discussion].

5. All other differences in the grouping are in low frequency events with small effect on the frequency of the initiating-event category.

The generic plant-population initiating-event data used in OPRA and summarized in Table 5.8 of that report were taken from the EPRI report[2] with some modifications:

a. OPRA did not include the short experience from the Indian Point 3 plant.

b. OPRA used plant data for "full years," and neglected reported experience from the last year if it covered less than half a year of operation.

BNL believes that the use of the entire EPRI[2] data without modifications is more consistent; however, the effect of these modifications is very small.

The Oconee plant-specific data in Table 5.8 of OPRA were derived from the operating experience of all three Oconee units, because of their similarity in design and operating characteristics. OPRA states that "the principal sources of operating histories of the three Oconee units were incident reports, LERs evaluations of operating experience and the allowable operating transients logs." The OPRA provides in Appendix B6 several tables of the operating transient logs, summarizing 120 unit-trip events for all three Oconee units, covering the period from the date of their effective service to the end of March 1980 -- a total of 17.5 plant-years. BNL has reviewed these trip summaries and, on the basis of the short description given for each one, has grouped them according to OPRA initiating-event categories. In Table 4.2, this grouping is compared with that of the OPRA[1] (Table 5.8 of OPRA) and EPRI NP-2230[2] for the three Oconee units. It is apparent that the OPRA used the EPRI categorization for the Oconee units, but modified it with information from the Oconee trip summaries given in Appendix B6 of the OPRA.

Table 4.2 shows that from the evidence from the unit trip summaries, 15 occurrences of partial loss of MFW (T_3) of the EPRI categorization were recategorized as turbine trip transients (T_1), and as loss-of-MFW transients (T_2). BNL agrees that the unit trip summaries support this change and made the same modification in its reevaluation. However, on the basis of the unit trip summaries, BNL considered that three events of loss of MFW could also be categorized as precursor events of the types T_{11}, T_{13} and a case of turbine-bypass-valve (TBV) failure. As will be discussed later, the frequencies derived for T_{11} and T_{13} in OPRA are consistent with the experience of one event in 19 reactor years. The TBV-failure transient was included by BNL in T_9 -- steam-line break.

The generic plant-population initiating-event data of the 35 PWRs were used by OPRA to generate a prior-event-frequency distribution for each of the transient categories. The Oconee operating histories were used to generate a posterior-event-frequency distribution by means of the Bayesian updating process. BNL has redone this calculation with its own code,[5] which uses a gamma-function event-frequency model as opposed to the OPRA code's lognormal event-frequency model. Comparison of the results in Table 4.3 shows a general agreement for the mean values and less of an agreement with respect to the uncertainty in the event frequencies, which is to be expected because of the different event-frequency distributions; it occurs primarily in cases in which strong evidence is lacking. One difference between the BNL and the OPRA results is the case of T_8. Since it has no effect on the core damage frequency, it was not further pursued by BNL.

4.1.2.2 Comparison With Other Studies

BNL has considered the Midland PRA,[3] the Arkansas-IREP,[4] and a B&W Owners Group report[6] in its comparison of initiating-event frequencies. Table 4.4 gives a comparison of all three sources, and the OPRA[1] and the BNL values. Some comments on the main differences follow:

a. The higher turbine trip frequencies for Arkansas and Midland are the results of using EPRI NP-2230 point estimates in the first case and one-stage Bayesian analysis in the latter case, whereas OPRA updated the generic EPRI data with Oconee plant-specific experience using two-stage Bayesian analysis; the Oconee plants experienced fewer turbine trips than the average PWR population. The posterior reflects the lower frequency of turbine trips (and transients in general) experienced by the Oconee units. Note that EPRI NP-2230[2] data reflect 9.7 transients per year for an average PWR vs 6.5 transients for the B&W plants. As seen from Table 2.14, the turbine-trip transients category in the Arkansas-IREP includes, in part, the case of a partial loss of MFW which in OPRA is considered part of another group.

b. The higher loss of MFW frequencies in the Arkansas and Midland are partly due to the inclusion of the loss of condenser vacuum transients in this category.

c. If the transients related to the interruption of the power conversion system (T_2, T_3, T_4, and T_7) are evaluated together, it can be seen that in OPRA and BNL their frequencies are higher than in MPRA, Arkansas-IREP, or the B&W Owners Group cases (e.g., 1.6 in OPRA vs 0.9 in MPRA). This is because EPRI transient No. 15 -- loss or reduction in FW flow (1 Loop), which is considered in OPRA as partial loss of FW, is included in the turbine trip category in the other PRAs.

d. Loss of offsite power in OPRA is plant specific, while it is evaluated generically in the other studies.

e. Excessive FW transient in OPRA is plant specific. It is smaller than the other studies because Oconee experienced only one event.

f. The B&W Owners Group study assesses a frequency of 0.052 for small steam-line breaks. MPRA groups TBV failures in the same category as steam-line breaks. On the basis of these, BNL included the one case of a TBV failure event identified in the Oconee trip summaries in this category, and obtained a frequency of 0.053; this may be conservative but, as shown in Section 5, its effect on the total core damage frequency calculated in this review is small.

g. On the basis of MPRA, the B&W Owners Group, and the fact that Oconee experienced events of ICS malfunctions in the past, a value corresponding to one event in 19 years was used by BNL for T_{11}. In assuming one event only, BNL gave credit to design changes made to rectify the past incidents.

h. Other significant differences are discussed in the next subsections on the quantification of special "rare-event" initiators.

4.1.2.3 Treatment of "Rare-Event" Initiators

OPRA derived the frequencies of some initiators in a special study considering additional sources of information beyond the EPRI NP-2230[2] and the plant trip summaries.

a. Pipe Break Initiators (T_9, T_{10}, A, S, and R)

Initiating-event categories T_9, T_{10}, A, S, and R are evaluated in OPRA on the basis of reported events of pipe breaks or leakages that occurred in one of the 35 plants considered in the OPRA Table 5.8. The experience considered in the OPRA is:

- T_9 One event of a rupture of a 6-in. steam line in H. R. Robinson Unit 2 in April 1970.
- T_{10} No event experienced in U.S. nuclear power plants.
- S One event that occurred in Zion Unit 1 in 1975.
- A No event experienced.
- R Three events of SG tube ruptures with leakage rates greater than 100 gpm: Surry Unit 2, November 1972, Point Beach Unit 1, February 1975, and Prairie Island Unit 1, October 1979.

A two-stage Bayesian analysis was applied to the above generic data and the Oconee plant-specific experience which reflects none of the above events in any of the three units.

BNL accepted the operating experience provided as basis for T_9, T_{10}, A, and R frequencies. For small LOCA(s), BNL considered that the Zion 1 event is applicable to the breaks ranging from 1.5 to 4 in. in diameter. For breaks smaller than 1.5 in., denoted in BNL review by the group "very small" or VS, BNL has considered the event of a seal failure in H. R. Robinson Unit 2 that occurred on May 1975, and added the frequency of 3.0×10^{-3} for the group of very small LOCAs that are not induced, but are caused by very small pipe breaks or by the spontaneous failure of the RCP seal itself.

MPRA has also used separate groups for the small and very small LOCAs (see Table 4.4). The very small LOCA in MPRA is derived from PWR experience given in the EPRI NP-2230 data for transient No. 6. This resulted in a mean value of 5.2×10^{-3}. However, the applicablity of this frequency estimation to the Oconee situation is not obvious. The B&W Owners Group report[6] provides a value of 8.3×10^{-3} stated to be due mainly to seal LOCAs and referred to the precursor study (NUREG/CR-2497).[7] The precursor study in Table C.1 (page C.8) shows that the H. R. Robinson event is responsible for 5×10^{-3}/yr out of the total of 8.3×10^{-3}/yr evaluated for very small LOCA (defined as breaks less than 1.5 in. in diameter).

In Table 4.3 the results of the two-stage Bayesian analysis performed in OPRA for these initators are shown. The set of LOCA frequencies used in the BNL review is given in Table 4.1.

b. Loss of Offsite Power (T_5)

The OPRA frequency of loss-of-offsite-power (LOOP) events was evaluated on the basis of EPRI NP-2301 (1982) study.[8] The data for the Southern Electric Reliability Council (SERC) Nuclear Power plants were used and include three LOOP events in 65 site-years. The Oconee site was assumed to have experienced one LOOP event (on 1/4/74). A two-stage Bayesian analysis was employed in OPRA using a gamma density function to represent the distribution of the LOOP frequency for each of the SERC plants. The analysis resulted in a mean annual frequency of 0.17 for LOOP at the Oconee site.

BNL reviewed this evaluation by using more recent data sources: NSAC-80 (1984) report[9] and NUREG-1032 (1985) draft.[10] They include 14 LOOP events for 105 SERC sites-years; both reports have almost identical lists of LOOP events. The small differences between the data given in these two studies are believed to result from further evaluation of the LERs of a few of the reported events. Using the more recent data for the SERC, assuming a gamma density function, and using the two-stage Bayesian methodology, BNL obtained a LOOP frequency of 0.12. The larger OPRA frequency is not due entirely to the differences in the data sources. The value 0.17 seems unreasonably high even if the older data (EPRI NP-2301) are used.

OPRA considered two initiating events in Category T_5:

1) T_{5FEED} -- loss of offsite power due to a failure of the grid or feeders, and
2) T_{5SUBF} -- loss of offsite power due to a substation failure.

The data for breaking T_5 into its constituents were obtained from the EPRI (1982) report,[8] which indicates that 78% of the LOOP events are substation type and 22% are grid related. This is close to the division used in OPRA. BNL used the more recent evaluations[9,10] for this breakdown too. A breakdown of 16 grid-related events and 30 substation-related events is given in the draft NUREG-1032.[10] The results are compared in Table 4.1.

The recovery data used in OPRA are given in Appendix D, pp. D-127 and D-130, and are derived from EPRI NP-2301.[8] BNL used the information from the two more recent sources[9,10] which are similar to each other. The derivation by BNL was based on its one-stage Bayesian code[5] using the Student-T distribution function. Table 4.5 compares the recovery data employed in OPRA with that used in the BNL review.

c. Loss of Instument Air (T_6)

OPRA derived the frequency of this initiator on the basis of the fault tree analysis of the instrument air (IA) system. BNL reviewed this analysis and agrees with the OPRA list of contributors as seen in Table 4.6. Some changes in quantification were made by BNL because of small changes in the failure rates, and because the factor 0.8 was used, rather than 0.7, to account for Unit 3 being at power during the fault.* Note that the value derived from the fault tree analysis is consistent with the occurrence of two loss-of-air events in the 11 system-years of operation.[11]

d. Loss of Low Pressure Service Water (T_{12})

OPRA provides a fault tree with various modules representing the different possible failure modes of the LPSW. BNL reviewed the analysis and agrees with the main contributors considered in OPRA. Small changes in quantifications were made as listed below. Table 4.7 compares the OPRA and BNL results for the evaluation of this initiator. The values for the pipe and valve breaks, the backwash, and other items were checked. The difference is seen to be due to the design that includes two additional valves[11]: CCW-94, suction from crossover (due to internal-flooding plant-design modification), and CCW-73, discharge manual value. This was partly balanced by the use of 0.8 (rather than 1.0) to account for Unit 3 being at power during the fault.

In both the OPRA and the BNL review, two initiators with different recovery probabilities were considered:

a) Failure of the LPSW: 3.5-3 (OPRA), 3.0-3 (BNL).
b) Failure of suction or discharge valves: 7.8-4 (OPRA), 1.9-3 (BNL).

e. Other "Rare" Initiating Events

a) Reactor Vessel Rupture (RPV): It was not reviewed in detail. The use of the UKAEA information, with credit given for in-service inspection, was judged to be reasonable and to result in a frequency consistent with other studies and other PRAs.

b) Loss of ICS Power Bus KI (T_{11}): BNL has increased the frequency of this initiator because of its judgment that although three similar events have occurred at Oconee the design changes later implemented would reduce the initiator frequency to the equivalent of about one event in the 19 years of plant operation. This results in a frequency of 0.05/yr. Note that one event occurred close to the end of the reporting period (in November 1979, while Appendix B.6 reports to March 1980).

*Based on Table 5.1 of the OPRA, excluding the TMI related shutdown period.

c) Spurious Low-Pressurizer-Pressure Signal (T_{13}): BNL has found the OPRA estimate of the mean annual frequency to be reasonable, on the basis of its judgment that one similar event was experienced (11/20/73). Considering the 19 reactor-years of experience at Oconee, the same frequency as in OPRA is obtained.

d) Loss of Power to 4-kV Switchgear 3TC (T_{14}): The frequency of this initiator remained unchanged in the BNL review. No such event in this particular bus has occurred in Oconee-3, and the failure rates used were accepted by BNL when the data were reviewed.

4.1.2.4 ATWS Initiators' Frequency

OPRA spent some effort evaluating the frequencies of ATWS initiators by collapsing the 41 transient categories of EPRI NP-2230 into 12 categories used in ATWS studies.[12] However, the information developed was not used in the final quantification of the ATWS initiator frequencies. It was used primarily for qualitative support for:

(a) Discarding some low frequency initiators, reducing the initiator categories from twelve to four (see Section 2.2.1.3).

(b) Claiming an approximately 20% margin in the results of the ATWS scoping study provided.

The frequencies finally used for ATWS initiating events were obtained from the derivation of the frequencies for the transient with scram as follows:

1. Turbine trip initiator frequency (including the partial loss of MFW and the excessive FW transients)	5.7
2. Loss of condenser vacuum	0.2
3. Loss of offsite power	0.2
4. Loss of MFW	0.7
Total	6.8

Since the total frequency of transients calculated in Section 4.1.1 was close to 7.0 per year, it is apparent that some part is missing. BNL judges that the loss-of-air transient is not accounted for in the above breakdown (T_6 = 0.17 per year). BNL included this transient in the loss-of-main-feedwater transient, and reduced the loss-of-offsite-power frequency to 0.12 to be consistent with the frequencies shown in Table 4.1.

BNL reviewed the information given in the OPRA Appendix E, and summarized it in its Table E.1 (Table E.1 in the OPRA has several misplaced headings). It was stated earlier that this information was used for qualitative support in the OPRA. BNL then reevaluated the information in EPRI NP-2230,[2] and used it to derive its own set of ATWS initiator frequencies.

The collapsing of the 41 EPRI NP-2230[2] transients into the 12 ATWS initiator for the Oconee units is not provided in OPRA. This was reconstructed by BNL and is shown in Table 4.8, in a format similar to Table E.1 of OPRA. The factor of frequency reduction due to consideration of power level greater than 25% was derived from this evaluation and used in the BNL frequencies.

1.	Turbine trip at power level >25% (including partial loss of MFW and excessive FW)	5.7*(4.3/5.1) = 4.81
2.	Loss of condenser vacuum	0.21*(0.2/0.25) = 0.17
3.	Loss of offsite power	0.12*1.0 = 0.12
4.	Loss of MFW (including T_8, T_9, T_{10}, T_{11}, T_{13}) (including the loss of air and assuming it has the same percentage at low power as loss of MFW)	0.87*(0.35/0.7) = 0.43
	Total	5.53

Assuming a benign consequence of transients occurring at power level less than 25%, the overall ATWS frequency is reduced by about 20%. However, the more demanding case of a loss of MFW is reduced even more.

4.2 Component Failure Data

The method used to develop the data base for the OPRA is the standard method used for plants with operating experience, i.e., Bayesian analysis to combine generic information obtained from industry experience, with plant-specific data. This method is considered to be state of the art in data base evaluation.

The OPRA provides in Appendix B the generic data used for practically all components in the fault trees; in this appendix, not only are the distributions given, but also the source from which they were obtained. BNL reviewed that appendix item by item to verify whether the data were correctly obtained, and found that a very good job was done in this respect.

The OPRA also provides summaries for plant-specific component-failure rate in Appendix B; in this part not all components from the generic data have plant-specific data. BNL checked against the LERs for valves[13] and pumps[14] and, since not all failures are reportable, found that in nearly all cases the OPRA specific failure data present more failures than the LERs data.

BNL did not check the Bayesian update process used in the OPRA. However, a comparison of the data used in the OPRA (the posterior distribution in the Bayesian analysis) with those used in several other studies[3,4,15] indicated that most of the failure probabilities used in the OPRA are similar to or larger than those used in recent PRAs.

In conclusion, it can be said that BNL accepted most of the data used in the OPRA, and very few changes were made; the only change that had some effect on core damage frequency was the failure of the circuit breaker to open on demand. This change was made because the generic data used in the OPRA come from the IEEE-500/1977,[16] and the data from the IEEE-500/1984[17] are much different from those in the previous edition. Accordingly, BNL conservatively used the upper bound from the OPRA specific data (~1.0E-3/d), which are the same as those used in the IREP study for ANO.[4]

4.3 Maintenance Data

The OPRA presents the analysis for maintenance unavailability (frequency and duration of maintenance) in Section 5.2 and Appendix B4.

A review of these sections indicated that the methodology (Bayesian analysis) used is acceptable. However, the analysis of the data used in the generic frequency and duration of the maintenance is beyond the scope of this review. Comparison of the maintenance unavailabilities used in the OPRA with those used in previous PRAs,[3,4,15] led to the following conclusions:

a) The OPRA, in general, has used higher maintenance unavailabilities than the ANO[4] and IP-3 PRAs.[15]

b) The OPRA, in general, has used lower maintenance unavailabilities than the Midland[3] and Seabrook PRAs.[18]

On the basis of the facts above, BNL accepted the maintenance unavailabilities used in the OPRA except for the maintenance unavailability for the Keowee hydro units and Lee gas turbine; BNL used the data from the Oconee experience without any updating. This modification has almost no effect on the core damage frequency, as can be seen in Appendix A.

4.4 References

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12. Analysis of B&W NSS Response to ATWS Events, BAW-1610, 1980.

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14. Sullivan, W. H. and Poloski, J. P., Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants, NUREG/CR-1205, Jan. 1980.

15. Indian Point Probabilistic Safety Assessment.

16. IEEE Std500 Nuclear Reliability Data Manual, The Institute of Electrical and Electronics Engineers, Inc., 1977.

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18. Seabrook PRA.

Table 4.1 Summary of Initiating Event Frequencies in OPRA and BNL Review

Initiator	Initiator Frequency Oconee	BNL	BNL Comments
T_1: Turbine trip	4.9	4.9	
T_2: Loss of MFW	0.64	0.50	Consistent with Oconee unit trip summaries
T_3: Partial loss of MFW	0.69	0.69	
T_4: Loss of condenser vacuum	0.21	0.21	
T_{5SUBF}: LOOP	0.13	0.08	More recent data source used by BNL*
T_{5FEEDF}: LOOP	0.04	0.04	More recent data source used by BNL*
T_6: Loss of air	0.17	0.21	See Table 4.6
T_7: Excessive FW	0.092	0.092	Consistent with one occurrence at Oconee
T_8: Spurious ESF	0.01	0.01	
T_9: SLB and TBV failure	0.003	0.053	One experienced TBV failure added by BNL
T_{10}: FW line break	9.-4	9.-4	
T_{11}: ICS power bus fails	0.02	0.05	One experienced event assumed by BNL
T_{12}: Loss of LPSW (total)	4.[illegible]	4.9-3	See Table 4.7
T_{12}(108): Loss of suction/discharge	7.8-4	1.[illegible]-3	See Table 4.7
T_{13}: Stuck open spray	0.044	0.044	Consistent with one event experienced
T_{14}: Loss of ac power to bus	5.4-3	5.4-3	
R: SG tube rupture	8.6-3	8.6-3	
RPV: RPV rupture	1.1-6	1.1-6	
A_{out}: Interfacing LOCA	1.4-7	1.4-7	
VS: Very small LOCA	-	3.0-3	Added by BNL based on one seal failure experienced in nuclear plants
S: Small LOCA	3.0-3	3.0-3	
A: Large LOCA	9.-4	9.-4	

*Also, different recovery data used in OPRA and BNL (see Table 4.5).

Table 4.2 Categorization of Experienced Events in the Oconee Plants With Respect to OPRA Initiating Events

Initiator Category	Oconee Unit Trip Summaries[1]				From EPRI-NP 2230	OPRA	BNL Review
	Unit 1	Unit 2	Unit 3	Total			
Turbine trip (T_1)	45	22	25	92	82	94	94
Loss of MFW (T_2)	3	2	1	6	9	13	10
Partial loss MFW (T_3)	5	4	1	10	27	12	12
Loss of condenser (T_4)	0	3	0	3	5	4	4
Excessive FW (T_7)	1	1	1	3	1	1	1
TBV failure (-)	0	0	1	1	-	-	1[4]
LOOP (T_5)	0	0	0	0	1	1	1
Loss of air (T_6)	0	0	0	0	-	0[2]	-
Spurious ESF (T_8)	0	0	0	0	0	0	0
ICS malfunction (T_{11})	1	1	1	3	0	0	1[5]
Loss of SWS (T_{12})	0	0	0	0	0	0	0
Stuck open spray (T_{13})	1	0	0	1	0	0	1
Loss of vital bus (T_{14})	1[3]	0	0	1[3]	0	0	0
Total events	57	33	30	120	125	125	125
Total years of operation				17.5	19.	19.	19.

[1]Based on BNL evaluation of the events reported in OPRA Appendix B6.
[2]BNL was informed in a meeting with Duke that one event occurred before 1980 and another one in 1984.
[3]Included in T_1 for frequency calculations. This loss-of-bus event was not in the 3TC bus. The 3TC is considered to be the most severe case.
[4]BNL included turbine bypass failure (TBV) in the category T9--Steam-Line Break.
[5]DPC made a design modification which makes the recurrence of past events unlikely.

Table 4.3 Oconee Updated Initiating-Event Frequencies (Calculated by Different Two-Stage Bayesian Codes)

Initiator Category		Initiating Event Frequency (yr^{-1}) Mean	5%	Median	95%
Turbine trip (T_1)	OPRA[1]	4.9	4.1	4.9	5.7
	BNL	4.8	3.5	4.4	5.5
Loss of feedwater (T_2)	OPRA	0.64	0.36	0.61	0.92
	BNL	0.53	0.28	0.47	0.73
Partial loss of FW (T_3)	OPRA	0.69	0.40	0.64	0.97
	BNL	0.80	0.45	0.72	1.1
Loss of condenser vacuum (T_4)	OPRA	0.21	0.083	0.18	3.8[2]
	BNL	0.23	0.086	0.20	0.38
Excessive feedwater (T_7)	OPRA	0.092	0.018	0.076	0.21
	BNL	0.11	0.021	0.088	0.24
Spurious ESF actuation (T_8)	OPRA	0.01[3]	7.8E-6	2.8E-3	0.043
	BNL	0.04	5.0E-3	0.030	0.084
Steam-line break (T_9)	OPRA	3.0E-3	1.0E-6	5.0E-4	1.2E-2
	BNL	4.6E-3	1.3E-4	2.4E-3	1.4E-2
Small LOCA (S)	OPRA	3.0E-3	1.0E-6	5.0E-4	1.2E-2
	BNL	4.6E-3	1.3E-4	2.4E-3	1.4E-2
Feedwater line break (T_{10})	OPRA	9.3E-4	6.9E-7	6.8E-5	2.8E-3
	BNL	6.9E-4	1.4E-6	5.0E-5	3.2E-3
Steam generator tube rupture (R)	OPRA	8.6E-3	2.6E-5	3.1E-3	2.7E-3
	BNL	1.3E-2	1.6E-3	1.0E-2	4.2E-2

[1]OPRA code applies the lognormal distribution. BNL code applies the Gamma distribution.
[2]Apparently, a typo.
[3]May be a typo. No significant effect on the core damage frequency.

Table 4.4 Comparison of OPRA and BNL Initiator Frequencies With Several Other Studies

Initiator	Arkansas IREP[1]	Midland PRA[3]	B/W Owner Group	OPRA	BNL Review
T_1: {Turbine trip	7.1	6.1	{4.1	{4.9	{4.9
{Reactor trip		1.9			
T_2: Loss of MFW	1.0	0.7	0.9	0.64	0.5
T_3: Partial loss of MFW				0.69	0.69
T_4: Loss of condenser vacuum				0.21	0.21
T_5: LOOP	0.32	0.135	0.14	0.17	0.12
T_6: Loss of air				0.17	0.21
T_7: {Excessive FW flow	-	0.22	0.22	0.092	0.092
{MUPS malfunction		2.9E-3			
T_8: Spurious ESF	-	-	-	0.01	0.01
T_9: SLB and TBV failure	-	6.7E-3	}0.052	3.-3	0.053
T_{10}: FW-line break	-	-	-	9.3E-4	9.3E-4
T_{11}: ICS malfunction	-	0.055	0.048	0.02	0.05
T_{12}: {Loss of SWS	2.6E-3	3.7E-6	-	4.0E-3	4.9E-3
{Loss of CCW	-	4.1E-5	-	-	-
T_{13}: Stuck open spray	-	-	-	0.044	0.044
T_{14}: {Loss of ac power bus	0.035	-	-	5.4E-3	5.4E-3
{Loss of dc power bus	0.036	-	-		
R: SG tube rupture	-	0.014	0.017	8.6E-3	8.6E-3
RPV: RPV rupture	-	-	-	1.1E-6	1.1E-6
Aout: Interfacing LOCA	-	7.7E-7	-	1.4E-7	
VS: Very small LOCA	0.020	5.0E-3	8.3E-3*	-	3.0E-3
S: Small LOCA	6.9E-4	3.3E-3	4.E-4	3.0E-3	3.0E-3
M: Medium LOCA	1.6E-4	4.7E-4	-	-	-
A: Large LOCA	8.7E-5	2.0E-4	- }	9.3E-4	}9.3E-4

*Taken from the ORNL precursor study NUREG/CR-2497. It includes induced LOCAs (about 40%).

Table 4.5 Loss-of-Offsite Power Recovery Data

Nonrecovery Probability	OPRA Substation	OPRA Grid	BNL Substation*	BNL Grid**
30-min nonrecovery	0.46	0.67	0.28	0.66
1-hr nonrecovery	-	-	0.19	0.48
2-hr nonrecovery	0.21	0.44	0.12	0.32
4-hr nonrecovery	-	-	0.07	0.18
8-hr nonrecovery	-	-	0.04	0.10
12-hr nonrecovery	0.07	0.22	0.03	0.06

*From Table A.4 of the Draft NUREG-1032, using BNL one-stage Bayesian computer code and Student-T probability distribution function.
**From Tables A.5 and A.7 of the Draft NUREG-1032, using BNL code as above.

Table 4.6 Loss of Instrument-Air-Initiator Frequency Contributors

Event	Description	Dominant Cut Set	Initiator Frequency (yr^{-1}) OPRA	BNL**
Contamination	Inadvertent IA system contamination with water or oil	AIAPICF	0.102	0.133
Pipe rupture	IA pipe rupture not repaired in 10 minutes*	AIAPILF* AIAPILIOF	0.052	0.059
Loss of SA and one IA train	Pipe leak in SA system and failure of one IA compressor to run	ASAPILF* AIACPCF*3	0.006	0.007
Loss of SA and one IA in maintenance	Pipe leak in SA system and one IA compressor in maintenance	ASAPILF* AIACPAM*3	0.003	0.004
Loss of SA and loss of RCW to IA	Pipe leak in SA system and RCW valve to IA fails closed	ASAPILF* ARCWIASVO*3	0.002	0.003
One IA train fails and SA interconnect and fails too	IA fails mechanically to run, and SA interconnect fails	AIACPCF*3* (ASAIAVDO + ASAIAVVH)	~0	0.001
Total			0.17	0.21

*The ability to repair or isolate a major leak in the IA system is complicated by the fact that the system was not included in the detailed design drawings -- which make the recovery operation more difficult. Some pipes and valves are not visible or accessible (OPRA, page A.15-10).

**Differences in the BNL reevaluation are due to BNL's use of a factor of 0.8, rather than 0.7, to account for unit 3 being at power during the fault, and to correctly use the failure data given on page A15-19 of OPRA.

Table 4.7 Loss of Low Pressure Service Water System -- Contributors to the Initiator Frequency

Event	Description	Initiator Frequency (yr^{-1}) OPRA	BNL	Comments
Pipe break in supply header	LPSW (or HPSW) suction break LPSW (or HPSW) valve break CCW crossover break	9.3E-4	~9E-4	From flooding study.
LPSW discharge failure	LPSW-108 or discharge manual valve transfer closed (CCW-73)	7.8E-4	1.3E-3	BNL considered the two valves. Power factor of 0.8 applied
Suction failure	CCW-94 valve transfers closed	-	6.2E-4	Result from CCW crossover isolation in the modified plant. Factor of 0.8 applied.
Backwash	Standby pump in backwash and running pump fails	7.5E-5	6.0E-5	BNL applied a factor of 0.8 for Oconee being at power.
Standby pump in maintenance	Standby pump in maintenance and running pump fails	4.0E-4	3.2E-4	As above.
Operator fails to start standby pump	Standby pump does not start and running pump fails	2.1E-3	1.7E-3	
Total		4.3E-3	4.9E-3	

Table 4.8 Mean Annual Frequencies of Transient Categories at Oconee (from EPRI-NP-2230)

Transient Category	EPRI-NP-2230 Grouping	All Power Levels		Power Level Greater than 25%	
		All years (19.8 years)	Subsequent years (16.8 years)	All years (19.8 years)	Subsequent years (16.8 years)
Loss of condenser vacuum	25,27,30	0.25	0.12	0.20	0.12
Turbine trip	3,9,12,15,19, 21,23,28,33, 34,36-40	5.10	4.40	4.30	3.70
Loss of main feedwater	16,22,24	0.70	0.48	0.35	0.18
Loss of offsite power	35	0.05	<0.01	0.05	<0.01
Load increase	26,29	0.05	0.06	0.0	<0.01
Loss of RCS flow	1,14	0.25	0.18	0.15	0.12
Control rod withdrawal	2	0.10	0.12	0.10	0.12
RCS depressurization	4,5,7	0.05	0.06	0.05	0.06
Boron dilution	11	<0.01	<0.01	<0.01	<0.01
Excessive cooldown	6,20	<0.01	<0.01	<0.01	<0.01
MSIV closure	17,18	N/A	N/A	N/A	N/A
Inactive RCS loop startup	13	<0.01	<0.01	<0.01	<0.01
Total		6.60	5.50	5.20	4.30

5. ACCIDENT SEQUENCE QUANTIFICATION

This section presents the quantification of the accident sequences in the OPRA.[1] Subsections 5.1 through 5.3 present an overview of the OPRA approach used to quantify the accident sequences initiated by transients, LOCAs, and SGTR, anticipated transients without scram (ATWS), and interfacing LOCAs, respectively; BNL comments and modifications are also discussed. Subsection 5.4 presents the results of the BNL review compared with the OPRA results; further details are provided in Appendices A and B. Subsection 5.5 presents the uncertainty analysis performed in the BNL review with comparisons with the OPRA, where possible.

5.1 Quantification Procedure for Transients, LOCAs, and SGTR

As discussed in Section 3, the OPRA constructed the following functional event trees:

- Transient.
- Small LOCAs.
- Large LOCAs.
- SGTR event tree.

For all the above event trees the same approach to quantification was used, and the main steps for performing the accident sequences quantification were:

1. Solution of system fault trees. In this step, the OPRA[1] used the SETS[2] computer code to find the minimal cut sets for each system. This step was performed to ensure that system-level fault trees were logically correct; the results of this step were not used in the accident sequence quantification. This review used the same method as the OPRA.

2. Construction and solution of fault trees for core damage bins. In this step the OPRA constructed core damage fault trees (CDFT) for each core damage bin (see Section 3.1.2 and Table 5.1); i.e., the functional event trees presented in Sections 3.1.4 and 3.1.5 were converted to core damage fault trees (CDFTs), and the supporting logic and system fault trees were used as input to these CDFTs. After constructing the CDFTs for each core damage bin, the OPRA used SETS[2] for quantification of the accident sequences for each bin.

   The only difference between the approach used in this review and that used in the OPRA was that this review quantified the CDFTs for each sequence instead of for each core damage bin, which allows for a more refined treatment of the success states in each sequence.

   In this step, both the OPRA and this review used screening probabilities for human errors, which allows for a more accurate evaluation of the human errors for each minimal cut set at the sequence level. Note also that no recoveries are yet considered in this step.

3. Review of results and iteration. This step, which is a must in any analysis, provides assurance that the results are consistent with the

models and data used in the quantification process, and was performed by both the OPRA and this review.

4. Quantification of final minimal cut sets. In this step, after verification that the results are consistent with the understanding of the actual behavior of the plant, the screening values used for human errors in step 2 are replaced by their best-estimate values. Also, in this step, consideration is given to operator actions to terminate sequences, and recovery factors are applied to each sequence minimal cut set to obtain the appropriate final core damage frequency. Both the OPRA and BNL performed this step, and differences appear in the quantification of some recovery actions, mainly due to the grace time available to the operators. In Appendix A, details of those differences are provided.

5.2 Quantification Procedure for ATWS

The quantification of the ATWS was done in the OPRA, and also in this review, in a simplified manner. The functional event trees were constructed (see Section 3 of this report for details), and probabilities/unavailabilities were estimated for each top event in the event tree. The assumptions used in the evaluation of these probabilities/unavailabilities are generally compatible with those used in the rest of the OPRA. Differences existed between BNL and the OPRA on some of the assumptions, and details of these differences are given in Appendix B of this report.

The sequences were quantified by direct multiplication of the branches in each sequence of the event tree; this quantification process was also used in this review.

5.3 Interfacing-Systems LOCA

In its Appendix F, the OPRA presents a detailed analysis of the interfacing-systems LOCA. The only system providing interfaces between low and high pressure piping that could result in a diversion of flow out of the reactor building is the low-pressure-injection system (LPIS). For this system, the following interfaces were considered.

1. LPI-system injection lines (two lines).
2. Low pressure auxiliary spray line.
3. The decay heat removal (DHR) suction line.

A detailed evaluation of all possible modes of failures that would cause an interfacing-systems LOCA through these lines was performed by the OPRA and reviewed by BNL. Since the frequency of the initiating events was very low, it was assumed in the OPRA that these frequencies represent core melt with containment bypass.

This review is in agreement with the models and failure rate/probabilities used in the OPRA, and for further details the reader should refer to the OPRA Appendix F.

5.4 BNL Review Results

This subsection presents the summary of the results obtained in this review with comparisons to the OPRA; for more details of similarities and differences between the results of this review and those in the OPRA, refer to Appendices A and B of this report.

The total frequency of core damage calculated in this review is equal to 9.3E-5/yr as compared to 5.4E-5/yr for the OPRA. In Tables 5.2 and 5.3, contributors to the core damage frequency are summarized by bin (see Table 5.1 for the definition of bins), and by initiating-event category; in both tables, a comparison with the OPRA is also given. From these two tables the following conclusions can be drawn:

1. The largest increase is present in Bin III (5.7E-5/yr in this review vs 3.0E-5/yr in the OPRA). The major contribution to this difference comes from events caused by loss of instrument air (as an initiator or due to loss of offsite power). This difference in CD frequency is mainly due to the assumption of time available for recovery of compressed air. In Oconee-3 a loss of instrument air causes the drainage of the upper storage tank (UST) to the condenser hotwell, because valve C-176, which is normally used as a means of hotwell makeup from the UST (hotwell level control system), fails open on loss of air. In the OPRA, between two and six hours were used for the quantification of the probability of failure to recover air. In a meeting held at DPC to discuss BNL comments regarding the OPRA, it was verified that about one hour would be more appropriate for the time for drainage of the UST into the hotwell, and therefore the time available for the operators to recover air or transfer the EFW suction from the UST (primary source) to the hotwell. This change modifies the probability of failure to recover air from the value used in the OPRA (5.5E-2) to the value used in this review (0.5), which is based on the OPRA assessment for recovery of air together with the judgment of the reviewers. For more details on specific sequences, see Appendix A.

   In this review, the loss of instrument air (T_6) becomes the most dominant contributor to the core damage frequency, being responsible for 49% of the CD frequency due to transients with scram and for 33% of the total. In the OPRA it is responsible for 11% of the CD frequency due to transients with scram and 6% of the total.

   Note that according to a Sept. 20, 1985 letter from H. B. Tucker (DPC) to H. R. Denton (NRC), Duke Power Company has taken an interim measure, i.e., closure of manual isolation block valve (C-175) upstream of air-operated valve C-175, to prevent the potential drainage of the upper surge tank (UST) following the loss of instrument air. In the near future, according to the referenced letter, a modification to air-operated valve C-176 is planned. If this modification is taken into consideration, the BNL-calculated core damage frequency for Bin III will decrease from 5.7E-5/yr to about 3.1E-5/yr and the total CD frequency will be equal to 6.7E-5/yr instead of 9.3E-5/yr.

2. In Table 5.3 it is also shown that the most important transient in the OPRA is the loss of low pressure service water transient (T_{12}), which is responsible for about 45% of the core damage frequency due to transients with scram (24% of the total CD). In this review, it accounts for 29% of the core damage due to transients with scram (19% of the total CD).

   It is explained in Appendix A that some degree of conservatism exists in this review because of the assumption that if valve CCW-73 fails closed a loss of LPSW to its most important loads, i.e., cooling of HPI and RCP pump motors and heat exchangers in the component cooling water system, will occur. However, a detailed pipe-flow calculation to verify this assumption is beyond the scope of this review.

   Note also that if a recent modification made in the discharge of the LPSW to the cooling of HPI pump motors (DPC Drawings Nos. PO-115B, Rev. 26, and PO-124D, Rev. 12) is considered, the contribution of the loss-of-low-pressure-service-water transient (T_{12}) to the total CD frequency will decrease from 1.8E-5/yr to about 4.0E-6/yr; i.e., the total CD frequency in this review will become 7.9E-5/yr instead of 9.3E-5/yr.

3. The differences in CD frequency for instrument air (T_6) and for loss of low pressure service water (T_{12}) accounts for 71% and 13% of the increase in the total CD frequency, respectively. The remainder of the difference in total CD frequency comes from small LOCAs (8%), ATWS (4%), and large LOCAs (3%).

4. In this review, the core damage frequency is larger than that in the OPRA for almost every bin.

5.5 Dominant Accident Sequences

The dominant accident sequences for each core damage bin is presented in Table 5.4; for definition of the core damage bins see Table 5.1. For comparison, the equivalent table in the OPRA (Table 8.1) is reproduced here as Table 5.5. In these tables, the dominant accident sequences are given by sequence type as discussed in detail in Appendix A. The most important sequences in Table 5.4 are described briefly below:

Sequence Type [F] T_6BU: Bin III, Frequency 2.9E-5/yr (OPRA = 4.7E-6/yr)

These sequences involve a loss of instrument air, as an initiating event, or as a result of loss of offsite power, or as a result of system faults after a reactor trip. Main feedwater is unavailable because of the loss of instrument air, and the emergency feedwater becomes unavailable if the steam driven pump is not available and air is not recovered, or if the operator fails to transfer the EFW steam-driven pump suction to the condenser hotwell and air is not recovered. After the loss of MFW and EFW, failure of the operators to establish HPI cooling and make feedwater to the SGs available from the Standby Shutdown Facility (SSF) will result in core damage. As discussed in detail in Appendix A, BNL and the OPRA differ primarily in the time available to the operators for recovery of instrument air. The OPRA assumes two to six hours for this recovery of air with a failure probability of 5.5E-2. In this review a time on the order of one hour is used, with a failure probability of 0.5, which is based on the fact that the OPRA analysis for recovery of air (OPRA Appendix C, page C-15/C-16) states that the dominant failure modes require considerable effort to recover, and the failure probability equal to 0.3 is given in the OPRA for failure to recover in two hours. The value used in this review is also partly based on the judgment of the reviewers.

These sequences account for about 31% of the total core damage frequency in this review, and for about 9% in the OPRA.

Note that if the modification described in Section 5.4 item 1 is considered, the core damage frequency for this sequence type will decrease from 2.9E-5/yr to about 3.0E-6/hr.

<u>Sequence Type [G] T_{12}BU: Bin III, Frequency 1.8E-5/yr (OPRA = 1.5E-5/yr)</u>

These sequences are characterized by failure of the LPSW as an initiator, or failure of the 4.6-kV bus 3TC with other failures in the second LPSW pump, or any other transient with failure of the LPSW. The loss of LPSW causes failure of the HPI pumps. Since in these sequences the RCPs are tripped, the failure of the HPI seal injection will result in a small RCS leak (see page A3-16, item 2, in the OPRA) with inability to make up if the SSF seal injection is not actuated in about 30 min; item 2, page A3-16, of the OPRA states that "seal leakage will result if injection flow is interrupted and the RCPs are tripped."

In this review, the automatic HPSW makeup to the LPSW cooling of the HPI pumps is considered for the cases in which the loss of LPSW is not due to the blockage of the LPSW discharge path from the HPI pumps cooling and the component cooling; in the OPRA, this modification to the plant was not taken into account.

It is also important to note that the first cut set given in A.5.7.2 (Appendix A), i.e., loss of LPSW to cooling of the motor HPI pumps and component cooling system due to failures in the discharge path, is considered to be conservative. However, in order to evaluate the correctness of this failure mode (which was also later found but not included by the OPRA; see footnote in OPRA, page A14-27), pipe-flow calculations would be necessary. Thus, BNL does include this failure in some of its sequences. At a meeting held at DPC, drawings were given to BNL and NRC to show that this failure mode and that included in the second cut set in Appendix A (A.5.7.2) would be eliminated by the modification of the discharge of the SW from the cooling of the HPI pumps. However, this modification was not analyzed in the base case of this review, to be consistent with the OPRA. Note that the analysis of the Oconee-3 as is, i.e., with the modification described above, would change the core damage frequency due to these sequences to be equal to about 4.0E-6/yr. As a consequence of this modification, the total core damage frequency will decrease from 9.3E-5/yr to 7.9E-5/yr. Note that in the remainder of this section, this modification is not considered.

These sequences account for about 19% of the total core damage frequency in this review, while they contribute about 28% in the OPRA.

Sequence Type [A] SY_SX_S: Bin I, Frequency 5.4E-6/yr (OPRA = 5.0E-6/yr)

These core damage sequences are characterized by a small-break LOCA, with successful HPI injection. The LOCA causes the actuation of the RBSS (even with the successful operation of the RBCS), and the operators fail to terminate its operation. HPR fails to be initiated upon depletion of the BWST inventory or fails during operation.

There is almost no difference between this review and the OPRA, and these sequences account for about 6% of the total core damage in this review and for about 9% in the OPRA.

Sequence Type [B] AX_A: Bin VI, Frequency 4.8E-6/yr (OPRA = 4.8 E-6/yr)

These sequences are characterized by a large-LOCA initiating event, with successful injection but failure of the low pressure recirculation (failure of initiation or hardware failures).

There is no difference between BNL and the OPRA, and these sequences for about 5% of the total core damage frequency in this review and for about 9% in the OPRA.

Sequence Type [E] $T_{10}BU$: Bin III, Frequency 4.8E-6/yr (OPRA = 4.8E-6/yr)

These sequences are characterized by a large feedwater- or condensate-line break which results in failure of main and emergency feedwater. Failure of the operators to provide other sources of feedwater and failure to establish HPI cooling result in core damage.

There is no difference between this review and the OPRA, and the sequences account for about 5% of the total core damage frequency in this review and for about 9% in the OPRA.

Sequence Type [A] $A\bar{U}X_A$: Bin VI, Frequency 3.6E-6/yr (OPRA = 3.3E-6/yr)

These sequences are characterized by a large-LOCA initiator, with successful injection but failure of low pressure recirculation. The low pressure recirculation fails because high flow develops during the recirculation phase (a failure mode in the OPRA), and the operators fail to throttle the flow. Following this failure to throttle, pump cavitation and failure can occur.

There is practically no difference between BNL and the OPRA, and these sequences account for about 4% of the total CD in this review, and for about 6% in the OPRA.

Note that from the minimum cut sets presented in Appendix A of this report (also present in Appendix D of the OPRA) a single failure exists attributable to the common power supply to the LPI valves 3LP-12 and 3LP-17. According to the OPRA, Duke Power Company had initiated a modification to provide separate power supplies to these valves. This modification which is necessary to satisfy the single failure criteria practically does not change the core damage frequency for these sequences.

Sequence Type ATWS: Bin V, Frequency 3.6E-6/yr (OPRA = 1.7E-6/yr)

These sequences, which involve a transient with failure to scram, are characterized by a large primary-system pressure, which may exceed 3900 psig, due to: a) moderator temperature coefficient larger than the 95% value, or b) failure of primary relief, i.e., failure of the PORV or any of the SRVs to open, or c) failure of the MFW and partial failure of the EFW system (failure of the steam-driven pump, or failure of one motor-driven pump, or delay in EFW initiation). In all these sequences, it is postulated that a primary-system rupture will result. Following the primary-system break a failure to inject borated water occurs, resulting in core damage. For all these sequences, a probability equal to 0.1 is used for failure of injection because of possible deformations in valves (this value given by the OPRA is accepted in this review). It is important to point out that B&W analysis[3] indicates that no deformation will occur at about 3900 psig.

The main difference between this review and the OPRA is due to the probability used for failure of the function "primary relief." As discussed in Appendix B, Section B.1, the OPRA uses a value of 2.0E-2/d, while in this review a value of 0.17/d is used. This value of 0.17 was obtained by assuming that for 20% of the time of an equilibrium cycle primary conditions (MTC, etc.) are such as to require PORV relief in order to avoid a peak pressure higher than 3900 psig; it was also considered that the PORV block valve in Oconee-3 has been closed for about 80% of the time the plant is in operation. Note that, as discussed in Appendix B, Section B.7, the value used in this review is subjected to judgment, and a sensitivity study of total CD to this value is performed.

These sequences account for about 4% of total CD frequency in this PRA (vs 3% in the OPRA).

Sequence Type ATWS: Bin VI, Frequency 3.4E-6/yr (OPRA = 1.5E-6/yr)

The only difference between these sequences and the ATWS sequences described above is that here, injection of borated water is successful but long-term cooling fails.

The differences between BNL and the OPRA are exactly the same as discussed above. These sequences account for less than 4% in this review and for about 3% in the OPRA.

Note that if the PORV block valve is open during all times the plant is in operation, as stated in a Nov. 26, 1985 letter from H. B. Tucker (DPC) to H. R. Denton (NRC), the BNL-calculated ATWS core damage frequency for Bins V and VI will be equal to 1.4E-6/yr and 1.2E-6/yr respectively.

5.6 ATWS Sensitivity Analysis

As discussed in more detail in Appendix B, a limited sensitivity analysis was performed for the following functions in the ATWS event trees.

5.6.1 Primary Relief Function

The OPRA states that late in their work it was found that Oconee-3 had been operating with the pressurizer PORV block valve closed for about 80% of the time. However, in the quantification of the ATWS sequences, this was not considered. Since the failure of the PORV to open affects the peak pressure, which also depends upon other parameters such as the moderator temperature and Doppler coefficients and power level, it was assumed in the base case of this review that for 20% of operation time, primary conditions (MTC, etc.) are such as to require PORV relief to avoid a peak pressure higher than 3900 psig. If this fraction of time is changed to 10% or 50%, the following is the impact on the total ATWS core damage frequency (CD):

- Base Case (20%) - CD = 7.7E-6/yr.
- 10% - CD = 5.4E-6/yr.
- 50% - CD = 1.5E-5/yr.

Note that in the reviewer's judgment, which is based on the reactivity coefficients for an equilibrium cycle for Oconee-3, the fraction of the time in which the peak pressure may be larger than 3900 psig will be smaller than that used in the base case (20%).

5.6.2 Failure to Inject Borated Water (RCS peak pressure <3500 psig)

The OPRA used a value of 10^{-3} for the probability of failure to inject borated water for the cases in which the peak pressure will be smaller than 3500 psig. In this review, the same value was used for cases in which the MFW system remains on line, and a value of 2.0E-2 was used otherwise.

If, for all cases above, a probability of failure equal to 5.0E-2 (a value used in the NRC ATWS[4] task force analysis) is used, the total ATWS core damage frequency will change from 7.7E-6/yr to 1.3E-5/yr.

Note again that the values used in the base case are the more appropriate values in the opinion of the reviewer.

5.6.3 Failure to Inject Borated Water and Failure of Long-Term Cooling (RCS peak pressure >3900 psig)

In the OPRA and in this review (base case), a value of 10^{-1} was used for probability of failure to inject borated water or the failure of long-term cooling for cases in which the peak pressure may exceed 3900 psig and a LOCA will result. These values were used because of a probability of deformation of valves in the injection paths. If these values are changed from 0.1 to 0.2, the total core damage frequency will change from 7.7E-6/yr to about 1.5E-5/yr.

5.7 Uncertainty Analyses

In Chapter 12, Section 12.3, the OPRA presents a quantitative analysis of input data uncertainties. The computer code SPASM[5] was used to propagate the basic-event distributions to obtain their contributions to distributions for the core sequence type, core damage bin, and total core damage frequency. To obtain these distributions, the OPRA used only the core damage sequences with a frequency greater than 1.0E-6/yr. The distributions for the total core damage and for CD for each bin include the internal as well as external events.

This review also performed an assessment of the uncertainties about the frequency of core damage for internal events only. The uncertainty, as in the OPRA, should be interpreted as being introduced by uncertainties in the values of the various parameters, given the modeling assumptions described in previous sections. BNL used the SAMPLE[6] code to propagate the uncertainties; the uncertainties in the initiator and basic events were quantified by fitting lognormal distributions to evaluate uncertainty measures (mean and variance). For each bin, the uncertainties in the frequency of core damage were quantified by using the accident sequences that account for 90% of the bin core damage; the same approach was used for the uncertainty in the total core damage frequency. The results of this analysis are given in Table 5.6 and Figure 5.1. A comparison with the OPRA results is not possible because in the OPRA the results for each bin and for the total CD uncertainty include both internal and external events; in this review only internal events are considered.

5.6 References

1. Oconee PRA - A Probabilistic Risk Assessment of Oconee Unit-3, NSAC/60, June 1984.

2. Worrell, R. B. and Stack, D. W., A SETS User's Manual for the Fault Tree Analyst, NUREG/CR-0465, Nov. 1978.

3. Analysis of B&W NSS Response to ATWS Events, BAW-1610, Jan. 1980.

4. Recommendation of ATWS Task Force: Enclosure D, NRC-staff.

5. Leverenz, F. L., SPASM, A Computer Code for Monte Carlo System Evaluation, EPRI NP-1685 (1981).

6. Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG/75-014), October 1975.

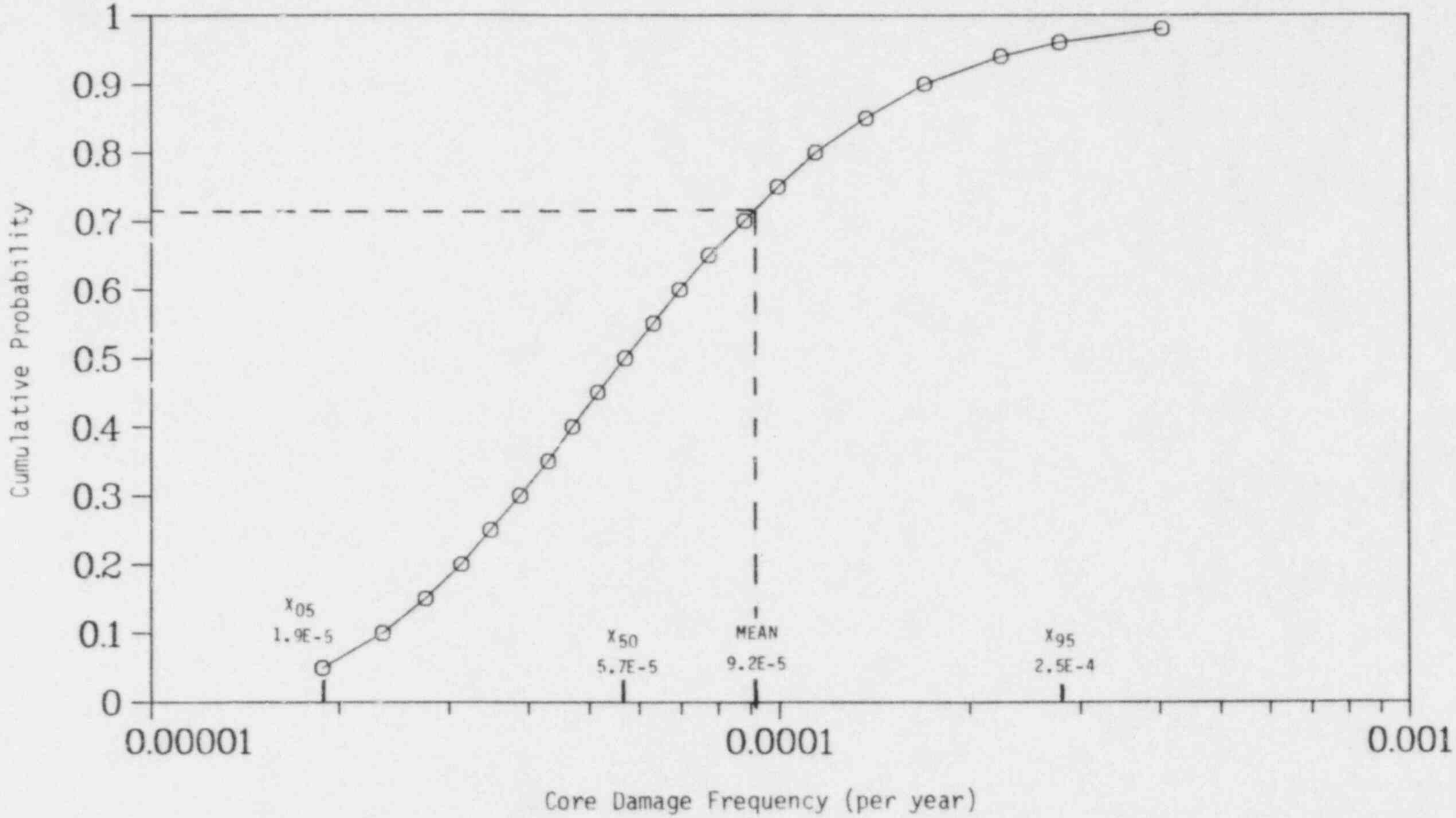


Figure 5.1 Cumulative probability for the frequency of core damage.

Table 5.1 Summary of Core Melt Bins

Bin	Sequence Characteristics
I	RCS pressure and leakage rates associated with small-break LOCAs, with early melting of the core (i.e., within about two hours after the break occurs)
II	RCS pressure and leakage rates associated with small-break LOCAs, with late melting of the core (i.e., after about 12 hours from when the break occurs)
III	High RCS pressure and leakage rates associated with boiloff of the reactor coolant through cycling pressurizer relief valves, with early core melting (within about two hours)
IV	High RCS pressure and leakage rates associated with boiloff of the reactor coolant through cycling relief valves, with late melting of the core
V	Large rates of leakage from the RCS and low pressures associated with large-break LOCAs with failure of core injection
VI	Large-break LOCA conditions with failure of coolant recirculation

Table 5.2 Summary of Contributors to Core Damage Frequency for Internal Initiating Events

CM Bin	Initiating Event	Core Damage Frequency OPRA	BNL
I	Pipe-break- and transient-induced small LOCA	6.5E-6	8.4E-6
	SGTR	1.3E-6	1.2E-6
	ATWS	1.8E-8	1.2E-8
	Total bin I	7.8E-6	9.6E-6
II	Pipe-break- and transient-induced small LOCA	1.1E-6	6.7E-6
	SGTR	1.4E-6	2.1E-6
	ATWS	1.8E-8	1.2E-8
	Total bin II	2.5E-6	8.8E-6
III	Transients	2.7E-5	5.7E-5
	ATWS	2.8E-6	6.9E-7
	Total bin III	3.0E-5	5.7E-5
IV	Transients	1.9E-7	3.6E-7
	Total bin IV	1.9E-7	3.6E-7
V	Large LOCA	1.4E-6	1.5E-6
	ATWS	1.7E-6	3.6E-6
	Total bin V	3.1E-6	5.1E-6
VI	Large LOCA	8.3E-6	8.5E-6
	ATWS	1.5E-6	3.4E-6
	Total bin VI	9.8E-6	1.2E-5
	Interfacing systems LOCA	1.4E-7	1.4E-7
	Total CD frequency	5.4E-5	9.3E-5

Table 5.3 Summary of Core Damage Frequency

Initiating-Event Category	Core Damage Frequency BNL	OPRA
Plant Transients		
Loss of Instrument Air (T_6)	3.1E-5	3.2E-6
Loss of Service Water (T_{12})	1.8E-5	1.3E-5
Feedwater Line Break (T_{10})	4.5E-6	4.8E-6
Loss of Offsite Power (T_5)	3.6E-6	2.4E-6
Loss of Reactor Trip (T)	1.7E-6	1.2E-6
Loss of Main Feedwater (T_2)	1.3E-6	1.2E-6
Other Transients	2.4E-6	2.6E-6
	6.2E-5	2.9E-5
Loss-of-Coolant Accidents		
Large Break (A)	1.0E-5	9.0E-6
Small Break (S)	9.2E-6[a]	6E.1-6
Reactor-Vessel Rupture	1.1E-6	1.1E-6
Total	2.0E-5	1.6E-5
Transients Without Scram (ATWS)	7.7E-6	6.0E-6
Steam Generator Tube Rupture (R)	3.3E-6	2.7E-6
Interfacing-System LOCA	1.4E-7	1.4E-7
Total	9.3E-5	5.4E-5

[a]Includes only LOCAs due to pipe breaks or spontaneous seal failures.

Table 5.4 BNL Review Summary of Core Damage Frequencies for Internal Events
- Total CD Frequency = 9.3E-5/yr.

Bin I Sequences			Bin II Sequences			Bin III Sequences			Bin IV Sequences			Bin V Sequences			Bin VI Sequences		
Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.	Type	Seq.	Mean Freq.
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0E-6 (ABOUT 89% OF TOTAL FREQUENCY)																	
													ATWS	3.6-6			
[A]	SY_SX_S	5.4-6*	[B]	T_6QX_S	3.2-6	[F]	T_6BU	2.9-5				[A]	VR	1.1-6	[B]	AX_A	4.8-6
			[C]	V_SX_S	1.9-6	[G]	$T_{12}BU$	1.8-5							[A]	AX_A	3.6-6
			[B]	RX_RO	1.5-6	[E]	$T_{10}BU$	4.8-6								ATWS	3.4-6
						[A]	T_2BU	1.3-6									
						[J]	TBU	1.1-6									
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0E-7 (ABOUT 99% OF TOTAL FREQUENCY)																	
[B]	RU_R	8.1-7	[C]	$S\bar{X}_S$	9.0-7	[C]	TBU	7.6-7		$T_{5,6}BLX$	2.3-7	[B]	AU	4.1-7			
							ATWS	6.9-7									
[B]	TQU_S	7.3-7	[A]	RX_RO	6.0-7	[B]	T_4BU	4.8-7		$T_{5,6}BLX$	1.3-7		I	1.4-7			
			[F]	TQX_S	3.7-7	[H]	$T_{5,6}BU$	2.5-7									
			[E]	TQX_S	2.2-7	[I]	$T\bar{B}U$	1.8-7									
			[A]	T_SQX_S	1.0-7	[D]	$T_{11}BU$	1.8-7									
[C]	SU_S	4.9-7															
[I]	TQU_S	4.8-7															
[C]	V_SU_S	4.2-7															
[F]	TQU_S	4.2-7															
[A]	RV_R	4.1-7															
[D]	T_6QU_S	2.9-7															
Total shown		9.6-6			8.8-6			5.6-5			3.6-7			5.2-6			1.2-5
Other		2.3-7			6.0-8			3.4-7			-			-			5.4-8
Total		9.5-6			8.8-6			5.7-5			3.6-7			5.1-6			1.2-5

*5.4-6 means 5.4E-6

Table 5.5 OPRA Summary of Core Damage Frequencies for Internal Initiating Events*

Bin I sequences			Bin II sequences			Bin III sequences			Bin IV sequences			Bin V sequences			Bin VI sequences		
Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.	Type	Seq.	Mean freq.
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0×10^{-6} (ABOUT 80% OF TOTAL FREQUENCY)																	
						[G]	$T_{12}BU$	1.5-5									
[A]	SY_SX_S	5.0-6**				[E]	$T_{10}BU$	4.8-6									
						[F]	T_6BU	4.7-6							[B]	AX_A	4.8-6
															[A]	AX_A	3.3-6
							TWS	2.8-6					TWS	1.7-6			
						[A]	T_2BU	1.2-6				[A]	VR	1.1-6		TWS	1.5-6
SEQUENCES WITH MEAN ANNUAL FREQUENCIES ABOVE 1.0×10^{-7} (ABOUT 95% OF TOTAL FREQUENCY)																	
[B]	RU_R	8.5-7															
			[B]	RX_RO	7.4-7												
			[C]	SX_S	6.9-7												
[B]	TQU_S	5.9-7															
[C]	SU_S	4.4-7	[A]	RX_RO	4.0-7	[C]	TBU	4.2-7									
[A]	RU_R	3.9-7				[B]	T_4BU	4.1-7									
[D]	T_6QU	2.5-7	[F]	$T_{8,13}QX$	2.2-7	[H]	$T_{5,6}BU$	2.2-7									
[I]	T_5QU	2.3-7							[B]	TBLX	1.6-7	[B]	AU	2.3-7			
[E]	$T_{12,14}QU_S$	1.0-7											I	1.4-7			
Total shown		7.9-6			2.1-6			3.0-5			1.6-7			3.0-6			9.6-6
Summation of others not shown		2.0-7			4.0-7			5.0-7			3.0-8			1.0-7			2.0-7
Total core-melt frequency		8.1-6			2.5-6			3.0-5			1.9-7			3.2-6			9.8-6

Notes: 1. The duplications of sequence types within a bin (e.g., two type [A] in bin I) are due to the sequences resulting from steam generator tube ruptures.

2. The sequences are defined by sequence type. This grouping of events considered (1) event-tree sequence type, (2) initiating-event effects if important, and (3) differences that would require unique treatment for the consequence analysis.

3. The TWS sequences are discussed in Appendix E. The line items in this table are summations of all TWS sequences for a bin.

4. Nomenclature: A, large LOCA; T, transient event; S, small LOCA; T_2, loss of main feedwater; T_5, loss of offsite power; T_6, loss of instrument air; T_8, spurious engineered-safeguard actuation; T_{10}, feedline break; T_{12}, loss of low-pressure service water; T_{13}, spurious low-pressurizer-pressure signal; T_{14}, loss of 4-kV switchgear 3TC; R, steam generator tube rupture; VR, reactor-vessel rupture; Q, loss of RCS integrity (transient-induced small-break LOCA); B, failure of RCS heat removal through the steam generators; U_T, failure of core heat removal by HPI cooling; U_S, failure of HPI in injection mode for small LOCAs; U_A, failure of LPI in injection for large LOCAs; U_R, failure of HPI in injection for tube ruptures; $Y_{T,S}$, failure to maintain RCS makeup supply; $X_{T,R,A,S}$, failure to maintain long-term core-heat removal appropriate to the initiating event; L, failure to recover RCS heat removal; O, failure to maintain long-term cooling at hot conditions for tube rupture; I, interfacing-system LOCA.

*Reproduced from OPRA, Table 8.1.

** 5.0-6 means 5.0E-6

Table 5.6 BNL Review Core Damage Frequency Distribution

Bin	X_{05}	X_{50}	Mean	X_{95}
I	8.4E-7	4.1E-6	8.8E-6	2.8E-5
IR	1.2E-7	6.8E-7	1.2E-6	3.9E-6
II	5.5E-7	3.0E-6	6.5E-6	2.2E-5
IIR	9.2E-8	7.9E-7	2.1E-6	7.2E-6
III	3.6E-6	2.2E-5	5.7E-5	1.9E-4
IV	3.1E-9	5.8E-8	3.5E-7	1.3E-6
V	4.8E-7	2.5E-6	5.1E-6	1.7E-5
VI	7.3E-7	4.9E-6	1.2E-5	4.0E-5

APPENDIX A

BNL REVIEW OF THE ACCIDENT SEQUENCES FOR TRANSIENTS, LOCAs, and SGTR

This appendix summarizes the detailed results of the BNL review of the OPRA internal events sequences. The methodology used to derive these sequences is basically the same as that used in the OPRA and is discussed in Sections 3 and 5 of the main report; the data used for the BNL accident sequence quantification are discussed in Section 4. For each comparison, this appendix is arranged in the same order as Appendix D, Section D2 through D4, of the OPRA.

For each sequence type, its corresponding event tree designation (Figures A.1 through A.4) is identified and a general description and frequency are provided. The corresponding number for the OPRA sequences is also printed in square brackets, and the containment-safeguard state is provided for each sequence. A list of the events occurring in each sequence, with its description and probability/unavailability, is provided in Table A.1.

A.1 BIN I SEQUENCE DESCRIPTION

The Bin I core damage sequences include small- and very-small-break LOCAs, both as initiating events and as transient-induced LOCAs, with early failure of core cooling due to (1) failure of the HPIS, or (2) failure to initiate high pressure recirculation (HPR) following depletion of the BWST inventory in about two hours.

A.1.1 Bin I, Sequence Type A, Event Tree Sequence SY_5X_5

These core damage sequences are characterized by a small-break LOCA (SLOCA), with successful HPI injection. The LOCA causes the actuation of the RBSS (even with the successful operation of the RBCS), and the operators fail to terminate its operations (YRBSH). HPR fails to be initiated upon depletion of the BWST inventory or fails during operation; the different modes of HPR failures are discussed in Subsection A.1.3.

A.1.1.1 Minimum Cut Set Listing - BNL CD frequency: 5.4E-6/yr
OPRA CD frequency: 5.0E-6/yr

BIN1A =

[7]	1	4.5000E-06	SLOCA * YBRSH * XHPR24 +
	2	2.2500E-07	SLOCA * YBRSH * SW7778CMH * RESW78 +
	3	1.5000E-07	SLOCA * YBRSH * LP40VVH * LP4142VVH +
[681,682, 682A]	4	1.5000E-07	SLOCA * YBRSH * IST-250 * IST-251 +
[130,131,183 469,470,521]	5	1.2696E-07	SLOCA * YBRSH * IST-247 * IST-249 +
[4]	6	5.0000E-08	SLOCA * YBRSH * LWDS3103VVH * RESUMPMF +
	7	4.3200E-08	SLOCA * YBRSH * IST-171 * IST-174 * RESWLPI +
	8	3.3120E-08	SLOCA * YBRSH * IST-174 * RESWLPI * IST-247 +
	9	3.3120E-08	SLOCA * YBRSH * IST-171 * RESWLPI * IST-249 +
	10	3.0000E-08	OTHERBIN1A +
[300]	11	4.1400E-09	SLOCA * YBRSH * IST-247 * IST-245 * LP14MVCH +
[299]	12	4.1400E-09	SLOCA * YBRSH * IST-249 * IST-244 * LP12MVCH

A.1.1.2 Discussion

The different modes of HPR failures are described below:

MCS 1 : Operators fail to attempt recirculation within 2 hr.
MCS 2,7: failure of LPSW to decay heat coolers.
MCS 3 : flow diversion from sump to BWST.
MCS 4 : failure of both sump line valves to be opened.
MCS 5 : failure of suction paths between the LPI decay heat cooler and the HPI pumps.
MCS 6 : failure of drain valves in suction line from sump, resulting in flooding of HPI pump room; operators fail to detect drainage and isolate.
MCS 8,9: failure of LPSW to decay heat cooler in one LPI train, and failure in the suction paths between the other LPI train and the HPI pumps.
MCS 11,12: failure of the suction path between one LPI train and the HPI pumps and failure in the other LPI train.

The main differences between BNL and the OPRA come from the addition of cut set number 3 and the change in the probability assigned for operator failure to correct the alignment of valves SW77 or SW78 (SW7778CMH).

A.1.1.3 Containment-Safeguard States

a) RBSS fails in the recirculation mode in all MCS, but No. 5.

b) No direct failures in MCS No. 5.

A.1.2 Bin I, Sequence Type B, Event-Tree Sequence TQU_5

These core damage sequences are characterized by a loss of LPSW with failure to recover. The loss of LPSW leads to a RCP seal failure and failure of the HPI pumps. If the RCPs are not tripped (HPRCPH), a leakage of about 100 gpm/pump will develop in about 1 hr; otherwise the leakage is only about 15 gpm.

A.1.2.1 Minimum Cut Sets Listing - BNL CD frequency: 7.3E-7/yr
OPRA CD frequency: 5.9E-7/yr

BIN1B =

	1	7.1500E-07	HPRCPH * T12E * RECCW73 +
[1]	2	4.7880E-09	T12C * HPRCPH * SWH247 * RESW12 +
[1]	3	4.2840E-09	T12A * HPRCPH * SWH247 * RESW12 +
[5]	4	3.8880E-09	HPRCPH * SWH247 * T14 * SWPAR * SW3BPPSH +
[188]	5	9.7920E-10	HPRCPH * SWH247 * SW3BPPSH * T * IST-173 +
[621]	6	3.4272E-11	HPRCPH * SWH247 * RESW12 * T * IST-173 * IST-175A

A.1.2.2 Discussion

In all MCS, the failure to trip the RCPs (HPRCPH) is present.

MCS No. 1, which does not appear in the OPRA, is due to a failure of the valve CCW-73 (T12E) and failure to recover (RECCW73). It is assumed, in this review, that this failure causes failure in the discharge of LPSW to the cooling in the HPI pump motors and to the component cooling system; this is probably a conservative assumption, but the realistic assessment of the effect of this failure is much beyond the charter of this review, and since it is not a very important contribution to the total CD frequency, it was not further evaluated. A realistic evaluation of this failure mode would require detailed pipe flow calculations.

All the other MCSs have a very low frequency in this review because the modification (after the analysis of the turbine building floodings) made to Oconee-3 in order to provide an automatic HPSW backup to the LPSW flow to the cooling of the HPI pump motors (SWH247 in MCS) was included. The OPRA did not include this modification.

A.1.2.3 Component-Safeguard State

a) RBCS and RBSS in MCS 2 through 6.

b) No direct failures in MCS No. 1.

A.1.3 Bin I, Sequence Type C, Event-Tree Sequences SU_S and V_SU_S

These sequences are characterized by a small- or very-small-break LOCA followed by failure of HPI.

A.1.3.1 Minimum Cut Set Listing - BNL CD frequency: 1.0E-6/yr
OPRA CD frequency: 4.4E-7/yr

BIN1C =

[100]	1	1.5000E-07	SLOCA * HP2425MVH +
	2	1.5000E-07	HP2425MVH * VSLOCA +
[13,198]	3	1.4400E-07	SLOCA * IST-186 +
	4	1.4400E-07	IST-186 * VSLOCA +
[121]	5	1.2288E-07	SLOCA * HP24MVO * HP25MVO +
	6	1.2288E-07	HP24MVO * HP25MVO * VSLOCA +
	7	3.8400E-08	SLOCA * HP25MVO * EACCHAMOD +
	8	3.8400E-08	SLOCA * HP24MVO * EACCHBMOD +
	9	8.5000E-08	OTHERBIN1C

A.1.3.2 Discussion

The failure of the HPI system in all these sequences is due to failure to establish suction flow to the HPI pumps from the BWST. The main difference between BNL and OPRA results is due to the fact that BNL has also considered a very small LOCA initiator (VSLOCA) (see Section 2 of this report).

A.1.3.3 Containment-Safeguard State

- Failure of RBSS for all sequences.

A.1.4 Bin I, Sequence Type D, Event Tree Sequence T_6QU_S

These sequences are characterized by a loss of instrument air (T_6), which causes loss of component cooling and loss of makeup flow to the letdown storage tank (LDST), followed by a failure to make the BWST available to the HPI pumps and failure to trip the RCPs.

A.1.4.1 Minimum Cut Sets Listing - BNL CD frequency: 2.9E-7/yr
OPRA CD frequency: 2.5E-7/yr

BIN1D =

[144]	1	1.0500E-07	HPRCPH * HP2425MVH * T6 +
[29,301]	2	1.0080E-07	HPRCPH * IST-186 * T6 +
[169]	3	8.6016E-08	HPRCPH * HP24MVD * HP25MVD * T6 +
[531]	4	3.7800E-10	HPRCPH * SWH247 * T6 * IST-172

A.1.4.2 Discussion

There is no difference between the OPRA and BNL review results.

A.1.4.3 Containment-Safeguard State

a) Failure of RBSS in MCS No. 2.

b) No direct failures in all other MCSs.

A.1.5 Bin I, Sequence Type E, Event Tree Sequence TQU_5

These sequences are characterized by a loss of LPSW, leading to loss of EFW and HPI. If MFW fails (SUMMFW or T_{14}), the SRVs will be challenged with water and with any one failing open (RCSRVLC) a LOCA is created; mitigation is lost because of loss of HPI.

A.1.5.1 Minimum Cut Sets Listing - BNL CD frequency: 5.5E-8/yr
OPRA CD frequency: 1.0E-7/yr

BIN1E =

[49,50,..]	1	3.4080E-08	SWH247 * QFDWR * RCSRVLC * IST-197 * T12 * SUMMFW +
[3,11,..]	2	8.7636E-09	SWH247 * T14 * SWPAR * QFDWR * IST-175 * RCSRVLC * IST-197 +
	3	1.2000E-08	OTHERBIN1E

A.1.5.2 Discussion

The major difference between BNL and OPRA results is due to the inclusion of the automatic HPSW backup to the cooling of HPI pump motors (SWH247) in the BNL review (see A.1.2.2).

A.1.5.3 Containment-Safeguard State

- Failure of RBCS and RBSS in all sequences.

A.1.6 Bin I, Sequence Type F, Event Tree Sequence TQU_5

These sequences are characterized by other losses of MFW events followed by failure of EFW and HPI, with a stuck-open SRV (RCSRVLC).

A.1.6.1 Minimum Cut Sets Listing - BNL CD frequency: 4.2E-7
OPRA CD frequency: 7.7E-8

BIN1F =

[12,27,..]	1	2.3625E-07	T6 * RCSRVLC * TREFWSUC * SUMHPI * REIA1 +
[12,27,..]	2	1.4490E-07	T6 * RCSRVLC * IST-197 * SUMHPI * REIA1 +
	3	1.5000E-08	OTHERBIN1F +
[132]	4	1.3500E-08	RCSRVLC * SUMHPI * T10 +
[31,45,..]	5	6.1200E-09	RCSRVLC * TREFWSUC * SUMHPI * T5SUBF * RESUBAIR1 +
[31,45,..]	6	3.7536E-09	RCSRVLC * IST-197 * SUMHPI * T5SUBF * RESUBAIR1

A.1.6.2 Discussion

Sequences 1, 2, 6, and 7 are initiated by a loss of instrument air (directly, T_6, or indirectly through LOOP, T_{5SUBF}) and failure to recover instrument air in one hour, REIA1 (time in which the upper storage tank will drain to the hotwell), followed by failure of the steam-driven EFW pump (IST-197), or failure to transfer EFW suction to the hotwell (TREFWSUC). With a stuck-open SRV (RCSRVLC) and failure of HPI (SUMHPI), core damage results. The main difference between BNL and the OPRA resides in the time available for recovery of instrument air (1 hr for BNL vs 2 to 6 hr in the OPRA); see Section 5 of this report.

Sequence 4 is initiated by a large feedwater line break (T_{10}) followed by a stuck open SRV and failure of HPI.

A.1.6.3 Containment-Safeguard State

- About 30% of all these sequences cause failure of the RBSS (because of the same suction path for HPI and RBSS).

A.1.7 Bin I, Sequence Type G, Event Tree Sequence $T_{13}QU_5$

This sequence is characterized by failure of pressurizer pressure control (T_{13}) resulting in stuck open SRV followed by failure of HPI.

A.1.7.1 Minimum Cut Sets Listing - BNL CD frequency: 6.3E-8
OPRA CD frequency: 6.8E-8

BIN1G =

[76,404,..]	1	6.3360E-08	SUMHPI * T13 * RCSRVSC

A.1.7.2 Discussion

There is no difference between the BNL review and the OPRA results.

A.1.7.3 Containment-Safeguard State

- About 30% includes the failure of the RBSS (see Subsection A.1.6.3).

A.1.8 Bin I, Sequence Type H, Event Tree Sequence $T_{10}Y_SX_S$

This sequence is characterized by a large feedwater line break (T_{10}) inside the reactor building (CPT10I), followed by the operators' failure to terminate the RBSS and to initiate the HPR in 2 hr (XHPR2H).

A.1.8.1 Minimum Cut Sets Listing - BNL CD frequency: 1.2E-8
OPRA CD frequency: 1.4E-8

BIN1H =

[324] 1 1.3500E-08 YBRSH * XHPR2H * RCSRVLC * T10 * CPT10I

A.1.8.2 Discussion

There is no difference between the BNL and the OPRA results.

A.1.8.3 Containment-Safeguard State

- Failure of the RBSS in recirculation.

A.1.9 Bin I, Sequence Type I, Event Tree Sequence TQU

These sequences are characterized by a blackout (P1R30, P1R1), failure of the steam-driven EFW pump (IST-197) on TREFWSUCM, failure to provide feedwater from the standby shutdown facility in 30 min after loss of the steam-driven EFW pump, and failure of the SRVs to close after liquid relief.

A.1.9.1 Minimum Cut Sets Listing - BNL CD frequency: 4.8E-7/yr
OPRA CD frequency: 8.1E-8/yr

BIN1I =

1 3.8000E-07 RCSRVLC * IST-197 * P1R30 +

2 4.9000E-08 RCSRVLC * TREFWSUCM * P1R1 * RESSFW30 +

3 4.7000E-08 RCSRVLC * TREFWSUCM * P1R1 * RESSFP

A.1.9.2 Discussion

The main differences between the BNL and the OPRA results are due to the following:

a) The frequency of a blackout in the BNL review is higher than that used by the OPRA (P1R30 = 4.2E-5/yr vs 1.4E-5/yr in OPRA) because of the inclusion of the failure of both main feeder buses following any initiating event, e.g., because of failure of the breaker, N1 and 3TC1 or 3TE1, or 3TD1 are included in the BNL review; these failure modes are not included in the OPRA.

b) The probability of failure of the operators to transfer the steam-driven EFW pump to the hotwell was also assessed to be larger in the BNL review (TREFWSUCM = 0.25 vs 0.15 in OPRA); this is due to the blackout conditions.

c) The probability of failure of the SSF backup power system, given a blackout (RESSFP = 0.096), was not included in the OPRA.

d) The timing for the EFW suction transfer is 1 hr in this review, as opposed to 2 hr. in the OPRA.

Note that the impact on the total core damage frequency is still very small, even with all these differences.

A.1.9.3 Containment-Safeguard State

- Failure of RBCS and RBSS.

A.1.10 Summary for Bin I Sequences

The total Bin I core damage frequency calculated in this review is equal to 8.4E-6/yr as compared to 6.5E-6/yr in the OPRA. The major contributions for this difference are: time for recovery of instrument and system, inclusion of very-small-LOCA initiator, changes in blackout frequency, and failure to transfer EFW suction to the hotwell.

A.2 BIN IR SEQUENCE DESCRIPTIONS

The Bin IR includes core damage sequences initiated by a steam-generator tube rupture (SGTR), followed by failure of the HPI.

A.2.1 Bin IR, Sequence Type A, Event Tree Sequences RBU_R

SGTR (R) with failure of HPI due to failure of the BWST to provide suction to the HPI pumps.

A.2.1.1 Minimum Cut Sets Listing - BNL CD frequency: 4.1E-7/yr; OPRA CD frequency: 3.9E-7/yr

```
                         BIN1RA =

[1,6]          1  4.12A0E-07   IST-186 * R
```

A.2.1.2 Discussion

There is no difference between this review and the OPRA.

A.2.1.3 Containment-Safeguard State

- Failure of the RBSS.

A.2.2 Bin IR, Sequence Type B, Event Tree Sequence $R\overline{B_R}U_R$

SGTR with failures of the HPI suction, other than the BWST.

A.2.2.1 Minimum Cut Sets Listing - BNL CD frequency: 8.1E-7
OPRA CD frequency: 8.5E-7

BIN1RB =

[4]	1	4.3000E-07	HP2425MVH * R +
[5]	2	3.5226E-07	HP24MVO * HP25MVO * R +
[23]	3	1.4310E-08	HP25MVO * R * HPSEGPM +
[24]	4	1.4310E-08	HP24MVO * R * HPSEGQM

A.2.2.2 Discussion

There is no difference between this review and the OPRA.

A.2.2.3 Containment-Safeguard State

- No direct failure of RBCS or RBSS.

A.2.3 Bin IR, Sequence Type C, Event Tree Sequence $R\overline{B_R}U_R$

SGTR with failure of HPI due to loss of LPSW.

There is no sequence of this type in the BNL review because of the inclusion of the automatic HPSW backup to the LPSW cooling of the HPI pumps.

A.2.4 Summary for Bin IR Sequences

The core damage frequency calculated in this review (1.2E-6/yr) is practically the same as in the OPRA (1.3E-6/yr). No differences were found.

A.3 BIN II SEQUENCE DESCRIPTIONS

These sequences involve a small and very small LOCA, or a transient-induced LOCA (transient with stuck-open SRV). The HPI is successfully actuated and the RBCS removes heat from the containment; the latter allows the BWST inventory to last longer than 12 hours. At the time of BWST depletion, HPR and LPR fail and core damage results.

A.3.1 Bin II, Sequence Type A, Event Tree Sequence $T_5\overline{QUY}X_5$

These sequences are initiated by a LOOP-initiating event causing a loss of instrument air, with failure to recover power to the instrument air in one hour (RESUBAIR1). A loss of EFW due to failures in the steam-driven EFW pump (IST-197) or to operator failure to transfer suction to the hotwell (TREFWSUC) will cause a challenge to the SRVs with one of them sticking open (RCSRVLC). HPI cooling and RBCS are successful. The recirculation with HPR or LPR fails because of the following: a) excessive LPSW to LPI coolers (SWEXCESS* SWEXCESSLPR), b) pump room flooding (LWD99103VVH), or c) operator failure to initiate HPR/LPR (XHPR12H).

A.3.1.1 Minimum Cut Sets Listing - BNL CD frequency: 1.0E-7/yr
OPRA CD frequency: 1.9E-8/yr

BIN2A =

	1	2.6400E-08	RCSRVLC * TREFWSUC * T5SUBF * RESUBAIR2 * SWEXCESS * SWEXCESSLPR +
	2	1.6192E-08	RCSRVLC * IST-197 * T5SUBF * RESUBAIR2 * SWEXCESS * SWEXCESSLPR +
[2A]	3	1.5840E-08	LWD99103VVH * RCSRVLC * TREFWSUC * T5SUBF * RESUBAIR2 +
	4	1.2240E-08	RCSRVLC * TREFWSUC * T5SUBF * RESUBAIR1 * XHPR12H +
	5	9.7149E-09	LWD99103VVH * RCSRVLC * IST-197 * T5SUBF * RESUBAIR2 +
	6	2.4000E-08	OTHERBIN2A

A.3.1.2 Discussion

The main differences between this review and the OPRA are due to the following:

a) The BNL consideration that, ven a loss of air, the air-operated valves downstream from the de ay heat coolers fail open and when the reactor building pressure reaches 3 psig the MOVs (MOV LPSW-4 and LPSW-5) open, resulting in excessive LPSW to the decay heat coolers. This excessive service water, according to the OPRA, may fail the coolers and therefore the HPR (SWEXCESS). It is also considered that the LPR may be disabled if several tubes are broken (SWEXCESSLPR). Note that in the fault trees given in the OPRA it was assumed that the excessive LPSW would disable the cooling function of the decay heat coolers with certainty. However, the DPC response to the BNL question explains why this assumption is not valid; BNL agrees with the reasons in that DPC answer.

b) The BNL assumption that if the operators fail to recover the instrument air in two hours no credit is given for possible actions to avoid pump room flooding due to the drainage of the reactor building sump (LWD99103WH).

A.3.1.3 Containment-Safeguard State

- Failure of RBSS.

A.3.2 Bin II, Sequence Type B, Event Tree Sequence $T_6\overline{QUYX_5}$

These sequences are very similar to the ones previously disabled in A.3.1. The only difference is that here the loss of instrument air is the transient initiator.

A.3.2.1 Minimum Cut Sets Listing - BNL CD frequency: 3.2E-6/yr
OPRA CD frequency: 1.1E-8/yr

BIN2B =

1	9.4500E-07	T6 * RCSRVLC * TREFWSUC * REIA2 * SWEXCESS * SWEXCESSLPR +
2	5.7960E-07	T6 * RCSRVLC * IST-197 * REIA2 * SWEXCESS * SWEXCESSLPR +
3	5.6700E-07	LWD99103VVH * T6 * RCSRVLC * TREFWSUC * REIA2 +
4	4.7250E-07	T6 * RCSRVLC * TREFWSUC * REIA1 * XHPR12H +
5	3.4776E-07	LWD99103VVH * T6 * RCSRVLC * IST-197 * REIA2 +
6	2.8980E-07	T6 * RCSRVLC * IST-197 * REIA1 * XHPR12H

A.3.2.2 Discussion

The differences discussed in Subsection A.3.1.2 also apply here.

A.3.2.3 Containment-Safeguard State

- Failure of RBSS.

A.3.3 Bin II, Sequence Type C, Event Tree Sequence $S\overline{UY}X_5$

These sequences involve a small- or very small-break LOCA followed by successful HPI injection. HPR and LPR fail because of a) failure to be initiated (XHPR12A), b) flow diversion to the BWST (LP40VVH*LP4142VVH), c) failure of sump valves to open (IST-250*IST-251), d) operator failure to stop the LPI pumps before their failure due to deadhead (LPPSTOPH) and failure to use the backup LPI pump C (LPIPUMPC), and e) pump room flooding (LWD99103VVH).

A.3.3.1 Minimum Cut Sets Listing - BNL CD frequency: 2.8E-6/yr
OPRA CD frequency: 6.9E-7/yr

BIN2C =

	1	9.0000E-07	VSLOCA * XHPR12H +
[85]	2	4.5000E-07	SLOCA * XHPR12H * /YBRSH +
	3	3.0000E-07	LP40VVH * LP4142VVH * VSLOCA +
	4	3.0000E-07	IST-250 * IST-251 * VSLOCA +
	5	2.4000E-07	VSLOCA * LPPPSTOPH * LPIPUMPC +
	6	1.8000E-07	LWD99103VVH * RESUMPMF * VSLOCA +
	7	1.5000E-07	SLOCA * LP40VVH * LP4142VVH * /YBRSH +
[A,B,C]	8	1.5000E-07	SLOCA * IST-250 * IST-251 * /YBRSH +
[23]	9	9.0000E-08	SLOCA * LWD99103VVH * RESUMPMF * /YBRSH +
	10	1.0000E-08	OTHERBIN2C

A.3.3.2 Discussion

The main difference between BNL and OPRA is due to the BNL inclusion of the very small LOCA initiator VSLOCA (see Section 2.3).

A.3.3.3 Containment-Safeguard State

- Failure of RBSS.

A.3.4 Bin II, Sequence Type D, Event Tree Sequence $TQ\overline{U}\overline{Y}X_5$

The dominant sequence involves a loss of feedwater initiated by a large feedwater break (T_{10}) followed by a stuck-open SRV (RCSRVLC) with successful HPI. When the BWST inventory is depleted, the operator fails to initiate HPR or LPR.

A.3.4.1 Minimum Cut Sets Listing - BNL CD frequency: 4.8E-8/yr
OPRA CD frequency: 4.8E-8/yr

BIN2D =

[356]	1	2.7000E-08	RCSRVLC * T10 * XHPLPR12H +
	2	2.0000E-08	OTHERBIN2D

A.3.4.2 Discussion

There is no difference between the BNL and the OPRA results.

A.3.4.3 Containment-Safeguard State

- Failure of RBSS.

A.3.5 Bin II, Sequence Type E, Event Tree Sequence $TQU\overline{YX}_5$

These sequences involve transients that lead to overcooling (T_9), and other transients (T) that are followed by additional failures (IST-255,I4007) also lead to overcooling. The actuation of HPI with operator failure to control injection (QHPIH), or open PORV block valve, RC417VCH (which is closed about 80% of the time), will challenge the SRV which sticks open after releasing liquid (RCSRVLL). When the BWST is depleted, HPR and LPR fail because of failure of initiation (XHPR12H), flow diversion (LP40VVH*LP4142VVH), or failure of the sump valves to open (IST-250*IST-251).

A.3.5.1 Minimum Cut Sets Listing - BNL CD frequency: 2.2E-7/yr
OPRA CD frequency: 5.8E-8/yr

BIN2E =

1	6.3600E-08	RCSRVLC * XHPR12H * T9 * RC417VCH * QHPIH +
2	3.0000E-08	OTHERBIN2E +
3	2.1200E-08	LP40VVH * LP4142VVH * RCSRVLC * T9 * RC417VCH * QHPIH +
4	2.1200E-08	IST-250 * IST-251 * RCSRVLC * T9 * RC417VCH * QHPIH +
5	5.5000E-08	T * RCSRVLC * XHPR12H * RC417VCH * QHPIH * IST-255 +
6	3.3000E-08	T * RCSRVLC * XHPR12H * RC417VCH * QHPIH * I4007

A.3.5.2 Discussion

These sequences do not appear in the OPRA because the PORV block valve was not considered to be closed for about 80% of the time. This was later realized by the OPRA team, but no modifications to the final results were made because of the small effect on total core damage frequency (see page D-77 of OPRA, Appendix D).

A.3.5.3 Containment-Safeguard State

- Failure of RBSS.

A.3.6 Bin II, Sequence Type F, Event Tree Sequence $TQU\overline{YX}_5$

These sequences involve primary-system pressurization by the pressurizer heaters (T_{13}), or inadvertent HPI actuation (T_8), or overcooling (T_9) followed by a stuck-open SRV relieving steam. The PORV is either failed (T_{13}) or blocked (RC417VCH). HPI is successful, and HPR and LPR fail for the reason given in Section A.3.5.

A.3.6.1 Minimum Cut Sets Listing - BNL CD frequency: 3.7E-7/yr
OPRA CD frequency: 2.2E-7/yr

BIN2F =

[258]	1	1.2672E-07	T13 * RCSRVSC * XHPR12H +
	2	4.2240E-08	LP40VVH * LP4142VVH * T13 * RCSRVSC +
[A]	3	4.2240E-08	IST-250 * IST-251 * T13 * RCSRVSC +
	4	4.0704E-08	LP40VVH * LP4142VVH * RCSRVSC * T9 * RC417VCH +
	5	4.0704E-08	IST-250 * IST-251 * RCSRVSC * T9 * RC417VCH +
[58]	6	2.5344E-08	LWD99103VVH * RESUMPMF * T13 * RCSRVSC +
	7	2.3040E-08	RCSRVSC * XHPR12H * RC417VCH * T8 +
	8	3.3000E-08	OTHERBIN2F

A.3.6.2 Discussion

The main difference between BNL and the OPRA is the inclusion of different failure modes for HPR and LPR.

A.3.6.3 Containment-Safeguard State

- Failure of RBSS.

A.3.7 Summary for Bin II Sequences

The total Bin II CD frequency in this review is equal to 6.7E-6/yr, as compared to 1.1E-6/yr in the OPRA. The major contributions to this difference are the inclusion of the very small LOCAs, the inclusion of the failure of HPR and LPR due to loss of air, and the different treatment for recovery of air (recovery time is different).

A.4 BIN IIR SEQUENCE DESCRIPTIONS

These sequences are characterized by a steam generator tube rupture (SGTR), with successful injection, and failure to establish a stable mode of long-term cooling before the BWST inventory is depleted.

A.4.1 Bin IIR, Sequence Type A, Event Tree Sequence $R\overline{B_R U_R}X_R O$

SGTR(R) with failure of a main steam relief valve on the affected SG to close (MSRVIC) such that HPR is not an option because the water to replenish the RPV is being discharged through the SG safety relief valve and the containment sump will be empty. The BWST fails to be refilled (OBWSTH) and decay heat removal function fails (XRDHRH, LPDHRSUC*REDHRSUC, or LPD16*RELPD16).

A.4.1.1 Minimum Cut Sets Listing - BNL CD frequency: 6.0E-7/yr
OPRA CD frequency: 4.0E-7/yr

BIN2RA =

[4]	1	3.4400E-07	R * OBWSTH * MSRV1C * LPDHRSUC * REDHRSUC +
	2	1.7200E-07	R * OBWSTH * MSRV1C * LPD16 * RELPD16 +
[61]	3	5.1600E-08	R * OBWSTH * MSRV1C * XRDHRH +
	4	3.2000E-08	OTHERBIN2RA

A.4.1.2 Discussion

The only difference between BNL and the OPRA is the inclusion of the failure of valves 3LP-19 and 3LP-20 (LPD16) in the BNL review.

A.4.1.3 Containment-Safeguard State

- Failure of RBSS.

A.4.2 Bin IIR, Sequence Type B, Event Tree Sequence $R\overline{B_R}\overline{U_R}X_RO$

SGTR with failure to refill the BWST before the inventory is depleted, and failure of the LPI system to operate.

A.4.2.1 Minimum Cut Sets Listing - BNL CD frequency: 1.5E-6/yr
OPRA CD frequency: 7.4E-7/yr

BIN2RB =

[1]	1	6.4500E-07	SW7778CMH * RESW78 * R * OBWSTH +
[18]	2	4.3000E-07	LP40VVH * LP4142VVH * R * OBWSTH +
	3	1.3000E-07	OTHERBIN2RB +
[11]	4	1.2384E-07	IST-171 * IST-174 * RESWLPI * R * OBWSTH +
	5	4.3000E-08	R * OBWSTH * RC660SVH * XRRCPH * XOLP1034H +
[70]	6	3.8700E-08	LP14MVCH * LP12MVCH * R * OBWSTH +
[30]	7	3.0960E-08	IST-171 * RESWLPI * LP14MVCH * R * OBWSTH +
[29]	8	3.0960E-08	IST-174 * RESWLPI * LP12MVCH * R * OBWSTH +
	9	2.5800E-08	R * OBWSTH * LPDHRSUC * REDHRSUC * OLTCH +
	10	1.2900E-08	R * OBWSTH * REDHRSUC * LPD16 * OLTCH +
[55]	11	1.0320E-08	IST-171 * RESWLPI * LP40VVH * R * OBWSTH

A.4.2.2 Discussion

The major difference between BNL and OPRA resides in the recovery of valves LPSW-77, or LPSW-78 (RESW78). In this review a probability of 5.0E-2 is used for the event RESW78, and a value of 5.0E-3 is used in the OPRA.

A.4.2.3 Containment-Safeguard State

- Failure of RBSS.

A.4.3 Summary of Bin IIR Sequences

The major contribution to the difference in core damage frequency between this review (2.1E-6/yr) and the OPRA (1.4E-6/yr) is the recovery of LPSW valves LPSW-77 or LPSW-78.

A.5 BIN III SEQUENCE DESCRIPTIONS

The sequences in this bin are characterized by transients in which the ability to remove decay heat via the steam generators is lost, and core cooling fails because the high-pressure injection cooling fails to be initiated, or is lost during the injection phase, or when the BWST is depleted in two hours (this last failure occurs only if the RBCS is lost).

A.5.1 Bin III, Sequence Type A, Event Tree Sequence T_2BU

Sequences with loss of MFW (T_2) initiating event followed by failure of EFW (EFUSTF, IST-197, IST-198,...) and failure of HPI cooling (UTHPIH).

A.5.1.1 Minimum Cut Sets Listing - BNL CD frequency: 1.3E-6/yr
OPRA CD frequency: 1.2E-6/yr

BIN3A =

[15]	1	6.9000E-07	T2 * UTHPIH * EFUSTF * REFDW2 +
	2	2.0000E-07	OTHERBIN3A +
	3	1.1730E-07	IST-197 * T2 * UTHPIH * REFDW2 * CCW87VVH +
	4	1.1730E-07	IST-197 * T2 * UTHPIH * REFDW2 * SW527VVH +
[105,168,425]	5	4.9680E-08	IST-197 * T2 * UTHPIH * REFDW2 * IST-198 +
[115]	6	4.8400E-08	T2 * UTHPIH * REFDW1 * IST-192 * IST-191 +
	7	4.8300E-08	IST-197 * T2 * UTHPIH * REFDW2 * SW606 +
[58,85]	8	4.6000E-08	IST-197 * T2 * UTHPIH * SWEFCCH * REFDW1 +
[105,168,425]	9	1.7940E-08	IST-197 * T2 * UTHPIH * REFDW2 * IST-199 +
[210]	10	1.1132E-08	IST-197 * T2 * UTHPIH * IST-177 * IST-176 * REFDW23

A.5.1.2 Discussion

There is essentially no difference between BNL and the OPRA.

A.5.1.3 Containment-Safeguard State

- No direct effects.

A.5.2 Bin III, Sequence Type B, Event Tree Sequence T_4BU

Sequences with loss of condenser vacuum initiator (T_4). The progression of these sequences is identical to the sequence type A above.

A.5.2.1 Minimum Cut Sets Listing - BNL CD frequency - 4.8E-7/yr OPRA CD frequency - 4.1E-7/yr

BIN3B =

[30]	1	2.8980E-07	UTHPIH * EFUSTF * REFDW2 * T4 +
	2	4.9266E-08	IST-197 * UTHPIH * REFDW2 * CCW87VVH * T4 +
	3	4.9266E-08	IST-197 * UTHPIH * REFDW2 * SW527VVH * T4 +
	4	2.0866E-08	IST-197 * UTHPIH * REFDW2 * IST-198 * T4 +
	5	2.0328E-08	UTHPIH * REFDW1 * IST-192 * IST-191 * T4 +
	6	2.0286E-08	IST-197 * UTHPIH * REFDW2 * SW606 * T4 +
	7	1.9320E-08	IST-197 * UTHPIH * SWEFCCH * REFDW1 * T4 +
	8	7.5348E-09	IST-197 * UTHPIH * REFDW2 * IST-199 * T4 +
	9	4.6754E-09	IST-197 * UTHPIH * IST-177 * IST-176 * REFDW23 * T4

A.5.2.2 Discussion

There is essentially no difference between BNL and the OPRA.

A.5.2.3 Containment Safeguard State

- No direct effects.

A.5.3 Bin III, Sequence Type C, Event Tree Sequence TBU

Sequences initiated by a turbine trip or another initiating event that does not disable MFW(T), followed by loss of MFW (SUMMFW) and EFW (EFUSTF, and others) and failure to recover feedwater, and failure to initiate HPI cooling (UTHPIH).

A.5.3.1 Minimum Cut Sets Listing - BNL CD frequency: 7.6E-7/yr
OPRA CD frequency: 4.2E-7/yr

BIN3C =

[ALL OPRA SEQ]	1	4.7191E-07	T * SUMMFW * UTHPIH * EFUSTF * REFDW2 +
	2	2.8728E-07	OTHERBIN3C

A.5.3.2 Discussion

The only difference between BNL and OPRA is in the sequences denoted by a OTHERBIN3C, where the other failures of the EFW system appear.

A.5.3.3 Containment-Safeguard State

- No direct failures

A.5.4 Bin III, Sequence Type D, Event Tree Sequence T_{11}BU

Loss-of-ICS-power-initiating event (T_{11}) causes loss of MFW. Loss of EFW (EFUSTF and others) and loss of HPI (UTHPIH) follow.

A.5.4.1 Minimum Cut Sets Listing - BNL CD frequency: 1.8E-7/yr
OPRA CD frequency: 6.0E-8/yr

BIN3D =

[102]	1	1.1500E-07	UTHPIH * EFUSTF * REFDW1 * T11 +
	2	7.0000E-08	OTHERBIN3D

A.5.4.2 Discussion

The differences between BNL and the OPRA are due to a large frequency used for T_{11} (5.0E-2/yr in this review vs 2.0E-2/yr in the OPRA; see Section 4) and to the inclusion of other failures of the EFW system (included in sequence named OTHERBIN3D).

A.5.4.3 Containment-Safeguard State

- No direct failures.

A.5.5 Bin III, Sequence Type E, Event Tree Sequence T_{10}BU

Large feedwater or condensate line break (T_{10}) which causes loss of MFW and EFW. Feedwater from other sources fails to be initiated (REFDW1), and HPI cooling fails (UTHPIH).

A.5.5.1 Minimum Cut Sets Listing - BNL CD frequency: 4.8E-6/yr
OPRA CD frequency: 4.8E-6/yr

BIN3E =

[14]	1	4.5000E-06	T10 * UTHPIH * REFDW1

A.5.5.2 Discussion

There is no difference between BNL and the OPRA.

A.5.5.3 Containment-Safeguard State

- No direct failures.

A.5.6 Bin III, Sequence Type F, Event Tree Sequence TBU

These sequences are characterized by a loss of instrument air, as an initiating event (T_6), as a result of LOOP (T5SUBF, TSFEEDF), or following a reactor turbine trip (T) with other failures (IST-3), and failure to recover instrument air in one hour (REIA1, REFEEDAIR1, RESUBAIR1). Loss of EFW (IST-197, TREFWSUC, EFUSTF), failure to initiate HPI cooling (UTHPIH) and failure to provide feedwater from the SSF (RESSFW30) results in core damage.

A.5.6.1 Minimum Cut Sets Listing - BNL CD frequency: 2.9E-5/yr
OPRA CD frequency: 4.7E-6/yr

BIN3F =

[2]	1	1.5750E-05	T6 * TREFWSUC * REIA1 * RESSFW30 * UTHPIH +
[4,9]	2	9.6600E	T6 * IST-197 * REIA1 * RESSFW30 * UTHPIH +
[34]	3	9.6600E-07	T6 * UTHPIH * EFUSTF +
[6]	4	4.0800E-07	TREFWSUC * T5SUBF * RESUBAIR1 * RESSFW30 * UTHPIH +
[52]	5	3.6800E-07	T5SUBF * UTHPIH * EFUSYF +
	6	3.0353E-07	T * TREFWSUC * REIA1 * RESSFW30 * UTHPIH * IST-3 +
[8,12]	7	2.5024E-07	IST-197 * T5SUBF * RESUBAIR1 * RESSFW30 * UTHPIH +
[21]	8	2.4675E-07	T6 * REIA1 * RESSFW30 * UTHPIH * EFSU3 * EFSUH * IST-207 +
	9	1.8616E-07	T * IST-197 * REIA1 * RESSFW30 * UTHPIH * IST-3 +
[51]	10	1.8400E-07	T5FEEDF * UTHPIH * EFUSTF +
[35]	11	1.0500E-07	T6 * REIA1 * RESSFW30 * UTHPIH * SW191VVHF +
[39]	12	8.4000E-08	T6 * REIA1 * RESSFW30 * UTHPIH * SWC89VVH +
[5]	13	7.8000E-08	TREFWSUC * RESSFW30 * T5FEEDF * REFEEDAIR1 * UTHPIH +
[642]	14	1.8616E-08	T * UTHPIH * EFUSTF * IST-3 +
[27]	15	9.5200E-09	T5SUBF * RESUBAIR1 * RESSFW30 * UTHPIH * SW138AVO

A.4.5.6.2 Discussion

The main differences between BNL and the OPRA are the fol:wing:

a) In the OPRA, a 2- to 6-hr time interval was allowed for recovery of the instrument air before drainage of the UST into the hotwell due to the opening of valve 3C-176. At a meeting held at DPC (BNL, DPC, and NRC), DPC responded to BNL questions by acknowledging that this time allowance was too long and one hour would be more appropriate. Accordingly, this review used the one-hour time (i.e., REIA1 = 0.5 vs. REIA2/6 = 5.5E-2 in the OPRA), which accounts for most of the difference in the CD frequency.

b) A second factor is the frequency of the initiating event T_6 (see Section 4.1.2.3); BNL uses 0.21/yr as opposed to 0.17 in the OPRA.

It is important to note that in the sequences initiated by LOOP due to substation failures (T5SUBF), BNL calculates a lower frequency of core damage because the initiator frequency is smaller (8.0E-2/yr vs 1.3E-1/yr in the OPRA), and the probability of failure to recover air is also smaller in the BNL review; the probability of failure to recover air is smaller because the failure to recover power is smaller in this review (see Section 4).

A.5.6.3 Containment-Safeguard State

- No direct failures of RBSS or RBCS.

A.5.7 Bin III, Sequence Type G, Event Tree Sequence TBU

These sequences are characterized by the failure of the LPSW as an initiator (T12A, T12C, T12D, T12E), or failure of the 4.6-kV bus 3TC (T14) with other failures in the second LPSW pump (SW3BPPSH, IST-175A), or any other transient (T) with failure of the LPSW (CCW73VVT, SW108VVT, IST-173* SW3BPPSH). The loss of LPSW causes failure of the HPI pumps. Since in these sequences the RCPs are tripped, the failure of the HPI seal injection will result in a small RCS leak (see page A3-16, item 2, in the OPRA) with inability to make up if the SSF seal injection is not actuated in about 30 min (RESSFSI); item 2, page A3-16, of the OPRA states that a seal leakage will result if injection flow is interrupted and the RCPs are tripped.

In this review, the automatic HPSW makeup to the LPSW cooling of the HPI pumps (SWH247) is considered for the cases in which the loss of LPSW is not due to the blockage of the LPSW discharge path from the HPI pumps cooling and the component cooling (failures other than T12D, T12E, SW1089VVT, and CCW73VVT); in the OPRA, this modification to the plant was not taken into account.

It is also important to note that the first cut set given in A.5.7.1 below, i.e., loss of LPSW to the cooling HPI pumps and component cooling system due to failures in the discharge path, is considered to be conservative. However, in order to evaluate the correctness of this failure mode (which was also found later by the OPRA, but not included; see footnote in OPRA, page A14-27), pipe flow calculations would be necessary. Thus, BNL includes it as one of its sequences.

At a meeting held at DPC, drawings were given to BNL and NRC to show that this failure mode and that included in the second cut set below were eliminated by modifying the discharge of the SW from the cooling of the HPI pumps. However, even though this modification was not analyzed in this review, to be consistent with the OPRA, it is expected that the first two cut sets and also cut sets 5 and 6 will no longer be valid for the present Oconee-3 (as built).

A.5.7.1 Minimum Cut Sets Listing - BNL CD frequency: 1.8E-5/yr
OPRA CD frequency: 1.5E-5/yr

BIN3G =

[1A]	1	7.1500E-06	T12E * RECCW73 * RESSFSI +
[1A]	2	7.1500E-06	T12D * RESW108 * RESSFSI +
[1]	3	3.0600E-06	T12A * SWH247 * RESSFSI +
[72]	4	1.3167E-07	RECCW73 * T * RESSFSI * CCW73VVT +
[72]	5	1.3167E-07	T * SW108VVT * RESW108 * RESSFSI +
	6	5.5814E-08	SWH247 * SW3BPPSH * T * IST-173 * RESSFSI +
[1]	7	4.7880E-08	T12C * SWH247 * RESW12 * RESSFSI +
	8	3.8880E-08	SWH247 * T14 * SWPAR * SW3BPPSH * RESSFSI +
	9	1.3608E-09	SWH247 * RESW12 * T14 * SWPAR * IST-175A * RESSFSI

A.5.7.2 Discussion

Even though the BNL total CD frequency for this sequence is not much different from that in the OPRA, significant differences exist in some of the contributors, as explained above (Subsection A.5.7). The following factors account for these differences: a) the inclusion of the valve CCW-73 (T12E and CCW73VVT), which increases the CD frequency; b) the inclusion of the automatic HPSW backup to the LPSW cooling of HPI pumps (SWH247), which decreases the CD frequency; and c) the decrease in the frequency of T12D (blockage of valve LPSW-108) which also decreases the CD frequency.

Note that if the modification in the discharge of the SW cooling to the HPI pumps is to be considered, the CD frequency for this sequence would be about 4.0E-6/yr.

A.5.7.3 Containment-Safeguard State

- Failure of RBCS and RBSS.

A.5.8 Bin III, Sequence Type H, Event Tree Sequence TBU

These sequences are initiated by a loss of instrument air (T6) or by a LOOP (T5SUBF, T5FEEDF) which results in loss of air. As a consequence of loss of air, the LDST makeup is lost. With flow unavailable from the BWST to the

HPI pumps (IST-186, HP245MVH, others), and since the RCPs are tripped, because of loss of component cooling, seal leakage results; if RCPs were not tripped, a seal failure would occur. Failure to protect the HPI pumps by cycling them (REHPPPCS) or failure to recover LDST makeup (by recovering air; RESUBAIR90), and failure to initiate makeup from the SSF (RESSFSI) before seal leakage occurs result in slow RCS leakage with no ability to make up, and core damage occurs.

A.5.8.1 Minimum Cut Sets Listing - BNL CD frequency: 2.5E-7/yr
OPRA CD frequency: 2.2E-7/yr

BIN3H =

	1	2.5600E-08	HP25MVO * T5SUBF * RESSFSI * REHPPPCS * HP24MVH +
	2	2.5600E-08	HP24MVO * T5SUBF * RESSFSI * REHPPPCS * HP25MVH +
[774]	3	2.0000E-08	HP2425MVH * T5SUBF * RESSFSI * REHPPPCS +
[272,833A]	4	1.9200E-08	IST-186 * T5SUBF * RESSFSI * REHPPPCS +
[833]	5	1.6384E-08	HP24MVO * HP25MVO * T5SUBF * RESSFSI * REHPPPCS +
	6	1.2800E-08	HP25MVO * T5FEEDF * RESSFSI * REHPPPCS * HP24MVH +
	7	1.2800E-08	HP24MVO * T5FEEDF * RESSFSI * REHPPPCS * HP25MVH +
	8	1.2800E-08	HP25MVO * T5SUBF * RESSFSI * HP24MVH * RESUBAIR90 +
	9	1.2800E-08	HP24MVO * T5SUBF * RESSFSI * HP25MVH * RESUBAIR90 +
	10	1.0000E-08	HP2425MVH * T5FEEDF * RESSFSI * REHPPPCS +
	11	1.0000E-08	HP2425MVH * T5SUBF * RESSFSI * RESUBAIR90 +
	12	7.0000E-08	OTHERBIN3H

A.5.8.2 Discussion

The BNL review and the OPRA differ primarily in the time available to recover the instrument air. DPC has acknowledged that the time available to recover air and consequently LDST makeup is about 90 min; BNL has used this time (RESUBAIR90) in its review. Since the probability of failure to recover air given a LOOP in this review is smaller than that used in the OPRA (RESUBAIR90 = 2.5E-2 vs RESUBAIR12 = 8.1E-3 as used in OPRA) and also the initiator frequency for T5SUBF is smaller in the BNL evaluation (0.08/yr vs 0.13/yr in the OPRA), the total core damage in both studies becomes almost the same.

It is important to note that sequences with loss of instrument air do not show up (i.e., their frequency of occurrence is included in the OTHERBIN3H), because in the case of a loss-of-air initiator the emergency procedures direct the operators to open the valve CC-8, thus providing component cooling and avoiding the trip of the RCPs and the subsequent seal leakage. A recovery factor for this action was included (probability of failure equal to 0.2), making the frequency of these sequences lower than 1.0E-8/yr.

A.5.8.3 Containment-Safeguard State

- No direct failure of RBCS or RBSS.

A.5.9 Bin III, Sequence Type I, Event Tree Sequence TBU

Sequences are characterized by a loss of all ac power for longer than four hours (P1R4), with successful secondary heat sink provided by the turbine driven EFW pump. This review assumes that in four hours the batteries are depleted and the failure to provide feedwater from the SSF (RESSFW30, RESSFP) results in core damage.

The sequence type also includes the sequences with loss of ac power for more than 12 hours (P1R12) with successful secondary heat sink provided by TDEFWP and later from the SSF. Failure of the SSF makeup function to provide seal injection in 30 min (RESSFSI) will result in gradual loss of RCS inventory, loss of primary to secondary heat transfer, and core damage.

A.5.9.1 Minimum Cut Sets Listing - BNL CD frequency: 1.8E-7/yr OPRA CD frequency: 2.6E-8/yr

BIN3I =

1	7.9787E-08	RESSFW30 * P1R4 */TREFWSUC * /IST-197
2	7.6592E-08	RESSFP * P1R4 * /TREFWSUC * /IST-197
3	2.6000E-08	RESSFSI * P1R12

A.5.9.2 Discussion

The main differences between BNL and the OPRA are due to the BNL assumption that the batteries will be depleted in four hours (the OPRA assumes battery depletion in 12 hours), and to the frequency of loss of all ac power (discussed in Section A.1.9.2).

A.5.9.3 Containment-Safeguard State

- Failure of RBCS and RBSS.

A.5.10 Bin III, Sequence Type J, Event Tree Sequence TBU

These sequences are characterized by: a) loss of all ac power for 30 min followed by failure of the steam-driven EFW pump (IST-197) and failure to provide feedwater flow for the SSF (RESSFW30, RESSFP), or b) loss of all ac power for about two hours with the steam-driven EFW pump operations, but with failure to transfer its suction from the UST to the hotwell (TREFWSUCM), and failure to provide feedwater to the SG from the SSF.

A.5.10.1 Minimum Cut Sets Listing - BNL CD frequency: 1.1E-6/yr
OPRA CD frequency: 2.9E-8/yr

BIN3J =

1 3.8640E-07 IST-197 * P1R30 * RESSFW30 +

2 3.7096E-07 IST-197 * P1R30 * RESSFP +

3 1.8750E-07 TREFWSUC * RESSFW30 * P1R2 +

4 1.8000E-07 TREFWSUC * RESSFP * P1R2

A.5.10.2 Discussion

The main differences between BNL and the OPRA come from the following:

a) In the OPRA a loss of all ac power for longer than two hours was assumed for sequences with loss of the steam-driven EFW pump (IST-197), and a 30-min time is used in this review. Note that it is stated in the OPRA that getting any source of feedwater after 30 min of loss of feedwater is ineffective. Also, in response to BNL questions, DPC agreed that a 30-min time period should be used.

b) In the OPRA the failure of the SSF power system (RESSFP), which must be used in the event of a blackout, was not considered.

c) In the BNL review, the probability of failure to transfer the EFW suction from the UST to the hotwell was increased for the case of a blackout.

A.5.10.3 Containment-Safeguard State

- Failure of RBCS and RBSS.

A.5.11 Other Sequences in BIN III

These sequences, which do not belong to any of the sequence types discussed above, were added by BNL and are described below:

1. T7*EFUSTF*REFDW2*UTHPIH 1.3E-7/yr.

2. T7*OTHEREFWF*REFDW2*UTHPIH 7.9E-8/yr.

These sequences (event tree sequence TBU) are initiated by excessive feedwater (T7) with successful trip of the MFW pumps (success event does not appear in the sequence), followed by loss of EFW (EFUSTF or other failures OTHEREFWF), failure to recover feedwater (RTFDW2), and failure to initiate HPI cooling (UTHPIH). These sequences are similar to sequences in types A and B.

3. T10*CPT10I*YBRSH*XHPR2H 1.3E-7/yr.

This sequence (event tree sequence $TB\bar{U}YLX$) is initiated by a large feedwater breaker (T_{10}) inside containment (CPT10I); feedwater is lost, and HPI cooling is successful. The RBSS is initiated and the operators fail to terminate its operation (YBRSH). After depletion of BWST in about two hours, the operators fail to initiate HPR (XHPR2N).

A.5.12 Summary of Bin III Sequences

The BNL core damage frequency for this bin was calculated to be equal to 5.6E-5/yr (vs 2.7E-5/yr in the OPRA). This difference is primarily due to the time available for recovery of instrument air (sequence type F); the BNL review used one hour and the OPRA used two to six hours.

A.6 BIN IV SEQUENCE DESCRIPTIONS

These sequences involve a transient followed by failure of all feedwater for 6 or 12 hours. HPI cooling is successful and long-term recovery of feedwater fails, requiring HPR to be initiated; failure of HPR results in core damage.

A.6.1 Bin IV, Sequence Type A, Event Tree Sequence $TB\bar{U}\bar{Y}\bar{W}LX$

These sequences are initiated by loss of instrument air, as an initiating event or due to LOOP, with failure to recover the instrument air for about six hours (REIA6). EFW is lost because of failure of the steam driven pump or failure to transfer suction to the hotwell (IST-197 or TREFWSUC). HPI cooling is successful initially, but fails in about six hours because of pump room flooding (LWD99103VVH). If the SSF fails to provide seal injection and feedwater to the secondary side within 30 min of the loss of HPI, core damage will result.

A.6.1.1 Minimum Cut Sets Listing - BNL CD frequency: 1.3E-7/yr
OPRA CD frequency: 3.2E-8/yr

BIN4A =

1	7.5600E-08	LWD99103VVH * T6 * TREFWSUC * RESSFSIW30 * REIA6 +
2	4.6167E-08	LWD99103VVH * T6 * IST-197 * RESSFSIW30 * REIA6 +
3	1.0000E-08	OTHERBIN4A

A.6.1.2 Discussion

The differences between BNL and the OPRA are explained below:

a) BNL uses a time of six hours for recovery of air because the HPI will be lost at that time, and feedwater and makeup from the SSF must be initiated within about 30 min after loss of HPI (see also OPRA, page D-27, Section D.2.6.1); the OPRA used 12 hours.

b) The failure of the SSF (RESSFIW30) used in the OPRA (0.2) is not correct because this is true only for a blackout. BNL uses a value equal to 0.1, which is the correct value for the sequences in this class.

Note that the sequences appearing in the OPRA are, in this review, part of the other sequences in Bin IV A (OTHERBIN4A)

A.6.1.3 Containment-Safeguard State

- Failure of RBSS.

A.6.2 Bin IV, Sequence Type B, Event Tree Sequence $\overline{TBUYW}LX$

These sequences are initiated by a loss of air, as an initiating event (T_6), or failure of LOOP (T5SUBF, T5FEEDF) with failure to recover the instrument air for 12 hours. The EFW is lost because of failure of the steam-driven pump or failure to transfer suction to the hotwell (IST-196, TROFWSUC). HPI cooling is successful, but high pressure recirculation fails because of excessive SW to the LPI coolers (SWEXCESS) due to loss of air (see Section A.2.1.2), eliminating the availability of high pressure recirculation (failure of LPI coolers). The SSF fails to provide seal injection and feedwater to the secondary side (RESSFIW30), and core damage will result.

A.6.2.1 Minimum Cut Sets Listing - BNL CD frequency: 2.3E-7/yr
OPRA CD frequency: 1.6E-7/yr

BIN4B =

```
1  7.5600E-08   T6 * TREFWSUC * SWEXCESS * RESSFSIW30 * REIA12 +

2  4.8000E-08   TREFWSUC * T5SUBF * SWEXCESS * RESSFSIW30 * RESUBAIR12 +

3  4.6368E-08   T6 * IST-197 * SWEXCESS * RESSFSIW30 * REIA12 +

4  2.9440E-08   IST-197 * T5SUBF * SWEXCESS * RESSFSIW30 * RESUBAIR12 +

5  3.0000E-08   OTHERBIN4B
```

A.6.2.2 Discussion

The main differences between BNL and the OPRA are:

a) BNL added the excessive service water to the LPI coolers (SWEXCESS) as a failure of the HPR.

b) BNL correctly used the failure of the SSF to provide makeup and feedwater; the OPRA includes the failure of the SSF power (RESSFP), which should be used only for a blackout.

c) The OPRA used the wrong probability for operation failure to initiate recirculation in 12 hours (XHPR12H = 3.0E-3 instead of 3.0E-4).

Note that Items b) and c) explain why the OPRA cut sets (with correct values) appear in this review as part of others (OTHERBIN4B).

A.6.3 Summary of Bin IV Sequences

The main difference between the core damage in this review (3.6E-7/yr) and that in the OPRA (1.9E-7/yr) is due to inclusion of excessive service water as a failure of the HPR.

A.7 BIN V SEQUENCE DESCRIPTION

These sequences are all large-break LOCAs that result in core damage due to failure of injection.

A.7.1 Bin V, Sequence Type A, Event Tree Sequence AU

This sequence is due to the failure of the reactor vessel, which causes the failure of mitigation. This review is in agreement with the OPRA.

1.1E-6 RPVRUPTURE

A.7.2 Bin V, Sequence Type B, Event Tree Sequence AU

These sequences are characterized by a large-break LOCA initiator, with failure of the LPI injection due to hardware or operator error.

A.7.2.1 Minimum Cut Sets Listing - BNL CD frequency: 4.3E-7/yr
OPRA CD frequency: 3.6E-7/yr

BIN5B =

	1	9.3000E-08	A * LPABH +
[10]	2	9.3000E-08	LP40VVH * LP4142VVH * A +
	3	8.3000E-08	OTHERBIN5B +
[7,27]	4	4.4640E-08	IST-186 * A +
[15]	5	3.9292E-08	A * IST-236 * IST-239 +
	6	1.8135E-08	A * IST-239 * IST-241 +
	7	1.8135E-08	A * IST-236 * IST-243 +
[23]	8	1.8135E-08	IST-244 * LP12MVCH * A * IST-236 +
[24]	9	1.8135E-08	IST-245 * LP14MVCH * A * IST-239

A.7.2.2 Discussion

The main difference between this review and the OPRA is the inclusion of MCS No. 1, i.e., operator fails the LPI system (LPABH) because of misdiagnosis (see OPRA, Appendix A, pages A2-17 and A2-52). The OPRA misses this sequence, which is part of their LPI fault tree (Figure A2-2 in the OPRA).

A.7.2.3 Containment-Safeguard State

• Failure of RBSS in about 20% of the total sequence type frequency.

A.7.3 Summary of Bin V Sequences

There is essentially no difference between the bin V core damage frequency in this review (1.5E-6/yr) and that of the OPRA (1.5E-6/yr).

A.8 BIN VI, SEQUENCE DESCRIPTIONS

The sequences in this bin are characterized by a large-break-LOCA initiator with successful injection but with failures in the recirculation phase.

A.8.1 Bin VI, Sequence Type A, Event Tree Sequence $A\bar{U}X_A$

Large-LOCA-initiating (A) event and successful injection, but high flow develops during recirculation in the low pressure recirculation (LPFLOWH) and either the operators fail to throttle, or the power to the valves is lost, or the valves fail to operate (P23XLF, P63XLF, LP12MVC*LP14MVC), resulting in pump initiation and failure.

A.8.1.1 Minimum Cut Sets Listing - BNL CD frequency: 3.6E-6/yr
OPRA CD frequency: 3.3E-6/yr

BIN6A =

[1]	1	2.7900E-06	A * LPFLOWH * LPTHROTTLE +
[4]	2	6.6030E-07	A * LPFLOWH * P23XLF +
[6]	3	9.3000E-08	A * LPFLOWH * P63XLF +
[21]	4	3.8093E-08	A * LPFLOWH * LP14MVC * LP12MVC

A.8.1.2 Discussion

The only difference between this review and the OPRA comes from the probability of failure of components included in P23XLF and P63XLF.

A.8.1.3 Containment-Safeguard State

• Failure of RBSS in the recirculation mode if the LPI spare pump is not brought to operation.

A.8.2 Bin VI, Sequence Type B, Event Tree Sequence $A\bar{U}X_A$

Large LOCA initator with successful injection but failure of the LPR due to failure of initiation or failures during the mission time.

A.8.1.1 <u>Minimum Cut Sets Listing - BNL CD frequency: 4.8E-6/yr</u>
<u>OPRA CD frequency: 4.8E-6/yr</u>

BIN6B =

[2]	1	4.6500E-06	A * XALPRH +
[OTHER]	2	9.3000E-08	IST-250 * IST-251 * A +
[36]	3	1.7707E-08	IST-173 * IST-175 * A +

A.8.2.2 <u>Discussion</u>

There is no difference between BNL and the OPRA.

A.8.2.3 <u>Containment-Safeguard State</u>

• Failure of RBSS in all sequences and failure of RBCS and RBSS in Sequence 3.

A.8.3 <u>Bin VI, Sequence Type C, Event Tree Sequence $A\overline{U}X_A$</u>

Large-LOCA initiation with successful injection and failure of LPR due to pump room flooding (LWD99103VVH) (drainage through valves LWD-99 abd LWD-103), with failure of the operators to isolate this drainage (RESUMPMF) in time (about six hours).

[3] 5.6E-8 A*LWD99103VVH*RESUMPMF

There is no difference between BNL and the OPRA, and for this sequence the RBSS is also failed.

A.8.4 <u>Summary of Bin VI Sequences</u>

The bin VI core damage in this review (8.5E-6/yr) is essentially the same as that in the OPRA (8.3E-6/yr).

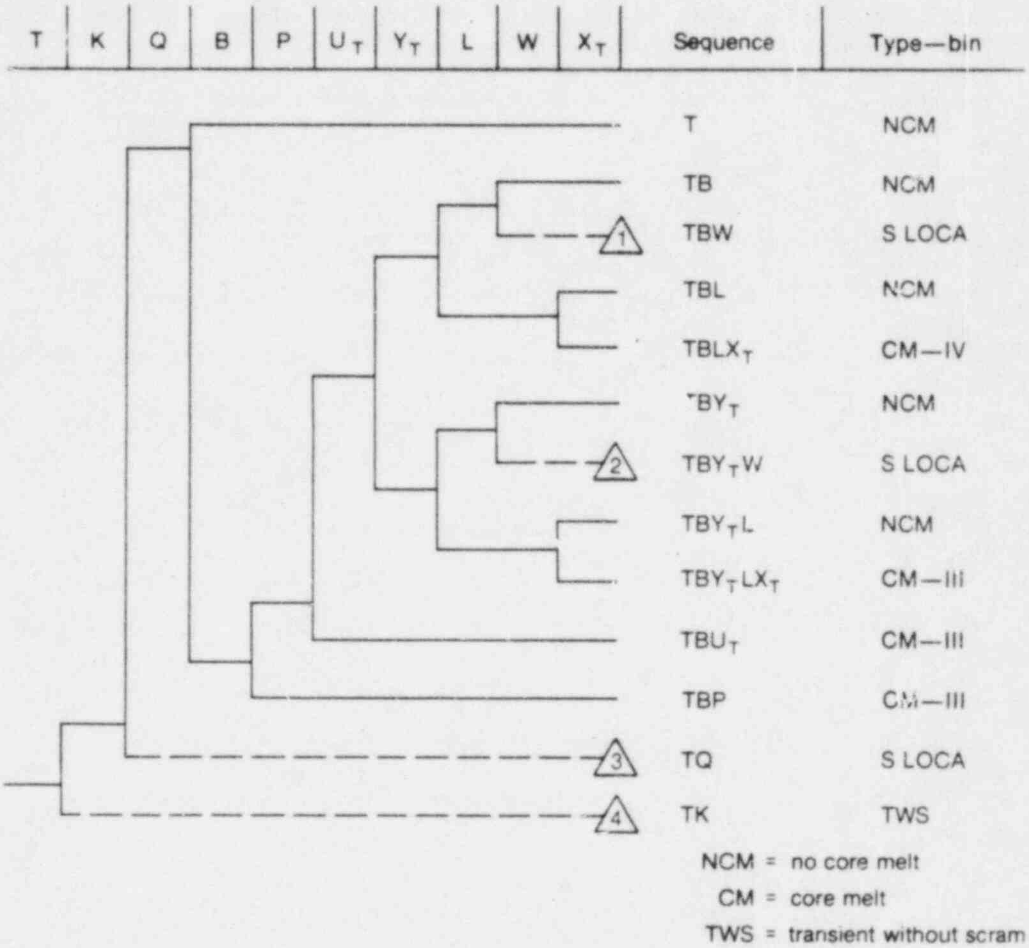


Figure A.1 OPRA event tree for transient-initiating events.

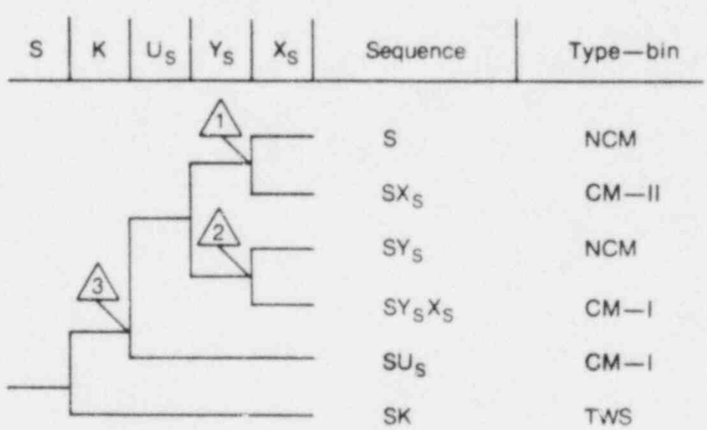


Figure A.2 OPRA event tree for small-break LOCA event.

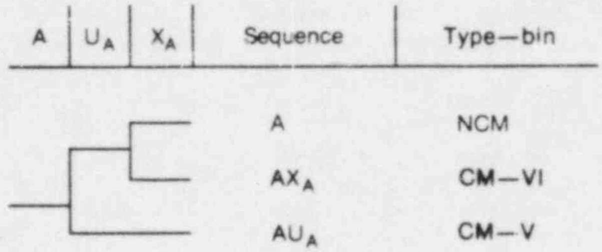


Figure A.3 OPRA event tree for large-break LOCA events.

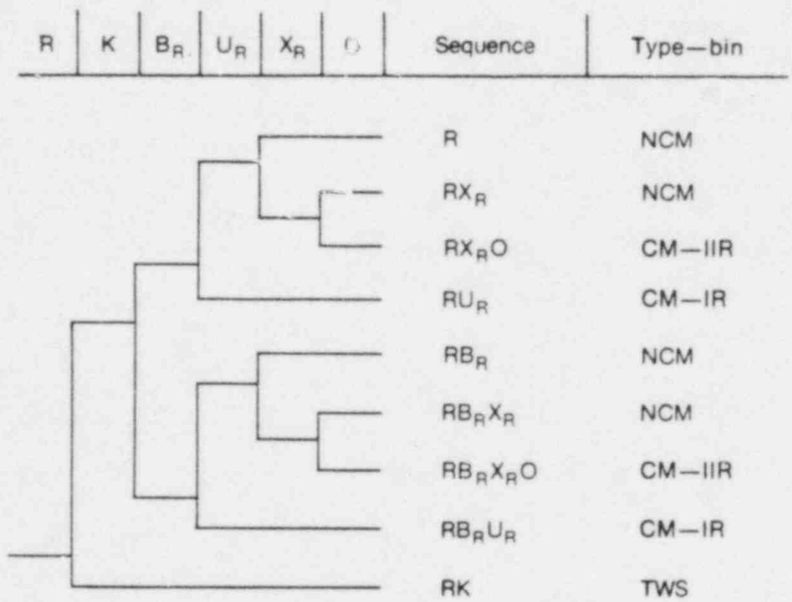


Figure A.4 OPRA event tree for SGTR initiating events.

Table A.1 Event Description Reference List

A	Large LOCA initiator	9.3E-4
CCW73VVT	CCW-73 transfers closed	2.1E-6
CCW87VVH	CCW-87 left closed	8.5E-4
CPT10I	Feedwater-line break inside containment	1.0E-1
EACCHAMOD	Failure of ES power supply to A channels	2.0E-3
EACCHBMOD	Failure of ES power supply to B channels	2.0E-3
EFSU3	Failure of startup steam supply from Unit 3	5.0E-1
EFSUH	Operator isolates steam to S/U header or loss of MFW without establishing alternative steam supply	1.0
EFUSTF	Insufficient water in the UST	4.6E-4
HP2425MVH	MOVs 3HP-24 and 3HP-25 left unavailable	5.0E-5
HP24MVH	Operator fails to open MOV 3HP-24	1.0E-2
HP24MVO	MOV 3HP-24 fails to open on demand	6.4E-3
HP25MVH	Operator fails to open MOV 3HP-25	1.0E-2
HP25MVO	MOV 3HP-25 fails to open on damand	6.4E-3
HPRCPH	Operator fails to trip RCPs on less of cooling flow	1.0E-2
HPSEGPM	Segment P (MOV 3HP-24, check valve 3HP-101) in maintenance	2.6E-4
HPSEGOM	Segment O (MOV 3HP-25, check valve 3HP-102) in maintenance	2.6E-4
LP12MVC	MOV 3LP-12 fails to throttle closed	6.4E-3
LP12MVCH	Operator inadvertently throttles valve 3LP-12 closed	3.0E-3
LP14MVC	MOV 3LP-14 fails throttle closed	6.4E-3
LP14MVCH	Operator inadvertently throttles valve 3LP-14 closed	3.0E-3
LP40VVH	Valve 3LP-40 left open	1.0E-3
LP4142VVH	Both valves 3LP-41 and 3LP-42 left open	1.0E-1
LP60VVO	Relief valve 3LP-60 fails to open	8.0E-3
LPABH	Operator inhibits/fails system	1.0E-4
LPDHRSUC	DHR suction flow from the reactor vessel is unavailable	2.0E-2
LPFLOWH	High flow (>4200 gpm) in A Loop (large LOCA)	1.0
LPIPUMPC	Operator fails to use LPI pump C given failure of pumps A to B	1.0E-1
LPPPSTOPH	Operator fails to stop pump for SLOCA during LPI	8.0E-4
LPTHROTTLE	Operator fails to throttle flow for Large LOCA	3.0E-3
LWD99103VVH	Drain valve not restored	6.0E-4
MSRVIC	One or more SRVs fails to close	4.0E-2
OBWSTH	Failure to initiate BWST refill in 12 hours	5.0E-1
OLTCH	Operator fails to cool at SG pressure	3.0E-3
P1R1	Failure of recovery power in 1 hour	1.7E-5
P1R2	Failure of recovery power in 2 hours	7.5E-6

Table A.1 Continued

P1R30	Failure of recovery power in 30 minutes	4.2E-5
P1R4	Failure of recovery power in 4 hours	1.1E-6
P3X9PF	Local failure of primary source to 3X8	6.1E-4
P23XLF	Local failure of 208-V MCC 3XL	7.1E-4
P23XDF	Local failure of 208-V MCC 3XD	7.1E-4
P63XLF	Local failure of 600-V MCC 3XL	1.0E-4
PFEED9CAF	Feed cable to 600-V MCC 3XS2 fails	6.0E-4
QFDWR	Failure to recover feedwater in 10 min. after loss of FW	7.0E-1
QHPIH	Operator fails to throttle HPI (on overcooling transients)	5.0E-2
R	Initiator SG tube rupture	8.6E-3
RC417VCH	PORV block valve left closed to inactivate PORV	8.0E-1
RC660SVH	Operator fails to open PORV when needed for HPI cooling	1.0E-2
RCSRVLC	Either SRV fails to close after liquid relief	1.0E-1
RCSRVSC	Either SRV fails to close after steam relief	9.6E-3
RECC8	Failure of open AOV 3CC-8 locally following closure as a result of loss of IA	2.0E-1
RECCW73	Failure of open valve 3CCW-73 locally	1.1E-1
REDHRSUC	Fails to open LPI suction MOVs for DHR, given failure of remote operation	1.0E-1
REFDW1	Failure to recover FW in 30 min.; one source available for recovery	5.0E-1
REFDW2	Failure to recover FW in 30 min.; two source available for recovery	3.0E-1
REFEEDAIR1	Failure to recover offsite power and reload IA within 1 hr. of T5FEEDF initiated	1.3E-2
REFEEDAIR90	Failure to recover offsite power and reload IA within 90 min. of T5FEEDF initiated	2.5E-2
REFEEDAIR12	Failure to recover offsite power and reload IA within 12 hrs. of T5FEEDF initiated	1.2E-3
REHPPPCS	Failure to operator to protect standby HPI pumps by allowing them to remain idle when suction unavailable	5.0E-2
REIA1	Failure to recover IA in 1 hour	5.0E-1
REIA90	Failure to recover IA in 90 minutes	4.0E-1
REIA2	Failure to recover IA in 2 hours	3.0E-1
REIA6	Failure to recover IA in 6 hours	1.0E-2
REIA12	Failure to recover IA in 12 hours	2.4E-3
RELPD16	Failure to recover the suction of RHR pumps	1.0E-1
RESSFP	Failure to initiate SSF due to failure of hardware (sequences involving blackout)	9.6E-2
RESSFSI	Failure to initiate SSF seal injection in 30 min. following a loss of seal injection	1.0E-1
RESSFW12	Failure to initiate FW from SSF within 12 hours	3.5E-3
RESSFW30	Failure to initiate FW from SSF within 30 minutes	1.0E-1
RESUBAIR1	Failure to recover offsite power and reload IA in 1 hour after substation failed	3.4E-2
RESUBAIR90	Failure to recover offsite power and reload IA in 90 minutes after substation failed	2.5E-2

Table A.1 Continued

RESUBAIR2	Failure to recover offsite power and reload IA in 2 hours after substation failed	
RESUBAIR12	Failure to recover offsite power and reload IA in 12 hours after substation failed	4.0E-3
RESUMPMF	Failure to find and isolate leakage from sump via LWD99 and 103 before HPI pump flooded	1.0E-1
RESW12	Failure to recover LPSW from Units 1 and 2	1.4E-2
RESW73	Failure to open valve CCW-73 locally	1.1E 1
RESW78	Failure to open valve SW-78 locally	5.0E-2
RESWLPI	Failure to recover failures that lead to isolation of LPSW to LPI cooler	2.0E-1
RESW108	Failure to open valve SW108 locally	1.1E-1
RPVRUPTURE	RPV rupture	1.1E-6
SLOCA	Initiator: small LOCA	3.0E-3
SW138AVO	AW-operated valve LPSW-138 fails to open	3.5E-3
SW108VVT	Valve transfer to close	2.1E-6
SW3BPPSH	Operator fails to start pump B	8.0E-3
SW527VVH	Valve LPSW-527 left closed	8.5E-4
SW606	Failures of valves LPSW-127 and LPSW-129	3.5E-4
SW778CMH	Valves LPSW-77 and 78 left in wrong position	3.0E-3
SWC89VVH	Manual valve CCW-89 left closed by operator	8.0E-4
SWEFCCH	Manual valves LPSW-513 and 518 inadvertently overthrottled	2.0E-4
SWEXCESS	Failure of HPR due to excess flow in LPI coolers	1.0E-2
SWEXCESSLPR	Failure of LPR given excess flow in LPI coolers	1.0E-1
SWH247	Failure of automatic HPSW backup to LPSW for HPI cooling	1.8E-2
SWPAR	Percent of time LPSW pump A is the operating pump	5.0E-1
SUMHPI	Failure of HPI due to hardware failure	1.5E-4
SUMMFW	Failure of MFW due to hardware failure	6.0E-2
T	The sum of iniators except loss feedwater	5.7
T1	Initiator: Reactor/Turbine Trip	4.9
T2	Initiator: Total loss of main feedwater	5.0E-1
T3	Initiator: Partial loss of main feedwater	0.69
T4	Initiator: Loss of Condenser Vacuum	2.1E-7
T5SUBF	Initiator: Failure of offsite power at the substation	8.0E-2
T5FEEDF	Initiator: Failure of electrical grid or main feeders	4.0E-2
T6	Initiator: Loss of instrument air	0.71
T7	Initiator: Excessive feedwater	9.2E-2
T8	Initiator: Spurious engineered-safeguards signal	1.0E-2
T9	Initiator: Steamline break	5.3E-2
T10	Initiator: Feedline break	9.0E-2

Table A.1 Continued

T11	Initiator: Loss of ICS power bus KI	5.0E-2
T12	Initiator: loss of service water	4.9E-3
T12A	Loss of LPSW (initiator) - Pump A fais and operator fails to start pump B	1.7E-3
T12C	Loss of LPSW (initiator) - suction failures	1.9E-3
T12D	Loss LPSW (initiator) - failure of valve LPSW-108	6.5E-4
T12E	Loss LPSW (initiator) - failure of valve CCW-73	6.5E-4
T13	Initiator: Spurious low-pressurizer-pressure signal	4.4E-2
T14	Initiator: Loss of power to bus 3TC	5.4E-3
TREFWSUC	Operator failure to transfer EFW suction from UST to hotwell	1.5E-1
TREFWSUCM	Same as above for blackout situations	2.5E-1
UTHPIH	Operator fails to attempt HPI cooling (feed and bleed)	1.0E-2
VSLOCA	Initiator: very small LOCA	3.0E-3
XALPRH	Operator fails to attempt LPR in 30 minutes	5.0E-3
XHPR2H	Operator fails to attempt HPR in 2 hours	3.0E-3
XHPR12H	Operator fails to attempt HPR in 12 hours	3.0E-4
XHPLPR12H	Operator fails to attempt LPR-HPR in 12 hours	3.0E-4
XOLP1034H	Operator fails to open LP-103 and LP-104	1.0E-1
XRDHRH	Failure to attempt cooldown	3.0E-4
XRRCPH	Operator fails to restart RCPs	1.0E-2
YBRSH	Operator fails to stop RB spray in 30 minutes given RBCS is operating	5.0E-1
IST-3	Loss of Instrument air during mission time	7.1E-4
AIAPICF		
AIAPILF		
AIAPILIOF		
AIAACVVT		
IST-171	No flow through LPI cooler A	1.2E-2
SW14MVO		
SW71VVH		
SW405AVT		
SW405AVH		
Others		

Table A.1 Continued

IST-172	No SW flow from LPSW header A to HPI pump cooling jacket	1.1E-5
SW148CVT		
SW147VVT		
SW347VVT		
SW347FLF		
IST-173	Failure of LPSW pump Train A	6.7E-4
SW3APPR		
Others		
IST-174	No flow through LPI cooler B	1.2E-2
SW05MVO		
SW72VVH		
SW404AVT		
SW404AVH		
Others		
IST-175	Failure of LPSW pump Train B	2.8E-3
SW3BPPSH		8.0E-3
SW3BFPH		1.4E-2
SW3BPPBM		3.0E-3
SW3BPPM		1.6E-3
Others		1.7E-3
IST-175A	Same as IST-175 without SW3BPPSH	2.0E-2
IST-176	No cooling flow through EF pump A	1.1E-2
SW517VVH		6.3E-3
SW516AVO		3.5E-3
SW509WH		8.0E-4
Others		2.8E-4
IST-177	No cooling flow through EF pump B	1.1E-2
SW526VVH		6.3E-3
SW525AVO		3.5E-3
SW518VVH		8.0E-4
Other		2.8E-4
IST-186	LPI loss suction	4.7E-5
LP28VVT		7.5E-6
LP28VVCH		2.8E-5
Others		1.2E-5

Table A.1 Continued

IST-191	The inlet of SG3A blocked	4.4E-3
EF200SVF		2.4E-3
EF315AVO		1.6E-3
Others		0.4E-4
IST-192	The inlet of SG3B blocked	4.4E-3
EF201SVF		2.4E-3
EF316AVO		1.6E-3
Others		4.4E-3
IST-197	TD pump fails to start or run	9.2E-2
EFTDPPS		3.8E-2
EFTDPPR		2.4E-2
EFTDPPIH		1.0E-2
SW137MVC		6.4E-3
EFTDPPM		5.2E-3
Others		8.0E-3
IST-198	MD EFW pump loss suction from UST	4.0E-4
CW573VVH		1.0E-4
CW166VVH		1.0E-4
CW180VVH		1.0E-4
CW572CVO		1.0E-4
IST-199	MD EFW pump loss suction from UST	1.0E-3
CW158MVH		1.0E-3
CW158MVT		4.3E-5
IST-207	TD EFW pump loss steam supply from SG3A & 3B	4.6E-3
MS87AVO		1.6E-3
MS86VVH		1.0E-3
MS90VVH		1.0E-3
MS89VVH		1.0E-3
MS91CVO		8.7E-5
IST-236	One of two LPI path blocked	6.5E-3
LP18MVO		6.4E-3
LP18MVMH		1.5E-4
IST-237	RHR suction path fails to openrun	6.9E-3
LP1MVO		6.4E-3
Other		5.4E-4

Table A.1 Continued

IST-241	LPI pump B fails to start and run	3.0E-3
LPBPPS		1.7E-3
LPBPPR		8.9E-4
LPBPPMH		4.5E-4
IST-243	LPI pump A fails to start and run	3.0E-3
LPAPPS		1.7E-3
LPAPPR		8.9E-4
LPAPPMH		6.5E-4
IST-244	Operator fails to reopen the discharge valve of LP pump train A after inadvertently closed	1.0E+0
LP12MVOH		1.0E+0
LP12MVO		6.4E-3
IST-245	Operator fails to reopen the discharge valve of LP pump train B after inadvertently closed	1.0E+0
LP14MVOH		1.0E+0
LP14MVO		6.4E-3
IST-247	Failure to deliver flow to HPI pump A suction due to valve closed during LPI-HPI mode	9.2E-3
LP15MVO		6.4E-3
LP15MVMH		1.8E-3
LP15MVH		1.0E-3
IST-249	Failure to deliver flow to HPI pump B suction due to valve closed during LPI-HPI mode	9.2E-3
LP16MVO		6.4E-3
LP16MVMH		1.8E-3
LP16MVH		1.0E-3
IST-250	Failure of the suction of LPR train B	1.0E-2
LP20MVO		6.4E-3
LP20MVRH		3.0E-3
Others		1.0E-3
IST-255	Two or more MSRVs open and fail to reclose or turbine fails to trip	8.0E-3
MSRVC		3.0E-3
MSTTF		5.0E-3
I4007	ICS failure causing loss of MFW (gate in ICS FT)	1.0E-2
LP19DHRH		5.0E-3
LP30DHRH		5.0E-3

APPENDIX B

BNL REVIEW OF SEQUENCES INVOLVING ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

B.1 INTRODUCTION

This appendix summarizes the results of the BNL review of the OPRA sequences involving anticipated transients without scram (ATWS). The OPRA presents only a scoping analysis of these accident sequences, and because of this the BNL review is also comparable to the analysis presented in the OPRA. The development of the ATWS accident sequences was discussed in Section 3 of this report, and the data used for quantification of these accident sequences will be discussed in this appendix. The order of the sequences appearing in this appendix is the same as that present in the OPRA, to provide an easy comparison.

Basically, this review agrees with the qualitative construction of the ATWS event trees (reproduced here in Figures B.1 through B.4). However, differences exist in the quantification. The main differences are the following:

1. The OPRA initiator frequencies for ATWS do not take into consideration the power level, and this review considers only the fraction of transients in which the power level exceeds 25% (see Section 4). This difference results in a decrease in core damage frequency in this review.

2. In its evaluation of the probability of the PORV to open, the OPRA fails to consider that the PORV block valve in Oconee-3 has been closed for about 80% of the time the plant is in operation. Since this affects the pressure spike, which is also dependent upon the reactivity coefficients and power levels at the time of the ATWS event, this review assumes that for 20% of the time during an equilibrium cycle, the failure of the PORV to open may result in a peak pressure higher than 3900 psig; due to the impact of this assumption on ATWS core damage frequency a sensitivity analysis on this parameter is performed. This difference between the OPRA and this review results in an increase in the core damage frequency for ATWS.

3. The OPRA makes a mistake when evaluating sequences 14 and 15 of Figure E-1 in Appendix E of the OPRA. A value of 0.1 is used for the top events "Borated Water" and "Long-term Cooling" (instead of 1.0E-3 as used in all equivalent sequences in the other event trees, and as discussed in the OPRA Appendix E). Correction of this error would bring down the total ATWS core damage in the OPRA from 6.0E-6/yr to about 3.6E-6/yr. In this review, a different probability is used for this event, as will be discussed below.

A detailed list of the ATWS sequences obtained in this review is given below. For each core damage bin, the most important sequences are presented with a short discussion, where differences between this review and the OPRA (Appendix D4 of OPRA) are presented. Note that in the quantification of the sequences presented below it is assumed that, for 20% of an equilibrium cycle, the failure of the PORV to open (because of valve failure or because the block valve is closed) creates a peak pressure greater than 3900 psig, i.e., the pressure at which B&W analysis (BAW-1600) indicates that some deformation of valves may occur. A sensitivity analysis to the value of this parameter is presented in a later subsection of this appendix.

B.2 BIN I ATWS SEQUENCES - BNL CD FREQUENCY: 1.2E-8/YR
OPRA CD FREQUENCY: 1.8E-8/YR

All sequences of this type involve an initiating event with failure to scram, followed by a small LOCA due to a stuck-open SRV with failure of boration. One sequence dominates this class:

ATWS sequence number 6 = 1.0E-8/yr (OPRA = 1.5E-8/yr).

The small difference between BNL and the OPRA is due to the frequency of initiating events (4.8/yr in this review vs 5.7/yr in the OPRA), and to the success state of the "primary relief" function. In the OPRA, a probability of 2.0E-2 is used for the failure of the function "primary relief," which includes the failure of the PORV (1.0-2/d) or any SRV to open (1.0E-2). This review takes into consideration the actual time that the PORV block valve is closed during operation (80%) and it is assumed that, for 20% of an equilibrium cycle, the failure of the PORV to open, or of any SRV to open at any time, is a failure of the function "primary relief." Thus, the failure of the function "primary relief" is given by

Percent of time that the failure of the PORV to open causes a peak pressure greater than 3,900 (base case: 20%)*

[Fraction of the time PORV block valve is closed (0.8) + Failure of PORV to open (1.0E-2)] +

Failure of any SRV to open (1.0E-2) = 1.7E-1/d.

B.3 BIN II ATWS SEQUENCES - BNL CD FREQUENCY: 1.2E-8/YR
OPRA CD FREQUENCY: 1.8E-8/YR

All sequences in this bin involve a transient initiator with failure to scram, followed by a small LOCA due to a stuck-open SRV with successful boration, but failure of long-term cooling. One sequence dominates this bin:

ATWS sequence number 5: 1.0E-8/yr (OPRA = 1.5E-8/yr).

The small difference between BNL and the OPRA is due to the same factors as explained in the previous subsection.

B.4 BIN III ATWS SEQUENCES - BNL CD FREQUENCY: 6.9E-7/YR
OPRA CD FREQUENCY: 2.8E-6/YR

All sequences in this bin involve a transient with failure to scram, followed by a failure to inject borated water, with successful heat removal through the SGs. The following are the dominant accident sequences:

ATWS sequence number 15 = 1.9E-7/yr (OPRA = 1.3E-6/yr).
ATWS sequence number 61 = 1.7E-7/yr (OPRA = 1.7E-8/yr).

ATWS sequence number 3 = 9.3E-8/yr (OPRA = 1.3E-7/yr).
ATWS sequence number 2 = 9.3E-8/yr (OPRA = 1.3E-7/yr).
ATWS sequence number 46 = 6.6E-8/yr (OPRA = 4.8E-9/yr).

The OPRA also presents as one of the dominant sequences the ATWS sequence number 14 with a frequency of 1.2E-6/yr.

The main differences between the BNL review and the OPRA are the following:

1. For the ATWS sequences numbers 14 and 15, the OPRA uses the wrong value for the failure to inject borated water and failure of long-term cooling. The value of 0.1, which in the OPRA is correct for the cases where a severe overpressure exists (peak pressure >3900 psig), is used instead of 10^{-3} which is used in the OPRA for cases in which the peak pressure is less than 3500 psig. In this review, for the cases in which the peak pressure is about 3500 psig and the EFW is supplying feedwater for the SGs, a value equal to 2.0E-2 is used (mainly operator failure to provide boration). For the case where the MFW is in operation, this review uses the same value as the OPRA (10^{-3}).

2. The frequency of the initiating events used for ATWS in the OPRA is bigger than that used in this review as discussed before (details are given in Section 4 of this report).

3. The failure of the "primary relief" function in the OPRA (2.0E-2) is much smaller than that used in this review (see Subsection B.2).

In conclusion, the most important difference between the CD frequency calculated in this review and that in the OPRA is due to item 1 above, i.e., due to an error in the OPRA. If correct values had been used for failure to inject borated water, the OPRA CD frequency for this bin would be equal to 3.3E-7/yr; this value is smaller than that obtained in this review.

B.5 BIN V ATWS SEQUENCES - BNL CD FREQUENCY: 3.6E-6/YR
OPRA CD FREQUENCY: 1.7E-6/YR

These sequences, which involve a transient with failure to scram, are characterized by a large primary-system pressure, which may exceed 3900 psig, due to: a) moderator temperature coefficient larger than the 95% value, or b) failure of primary relief, i.e., failure of the PORV or any of the SRVs to open, or c) failure of the MFW and partial failure of the EFW system (failure of the steam-driven pump or failure of one motor-driven pump). In all these sequences, a resulting primary-system rupture is postulated. Following the primary-system break, the failure of injection of borated water occurs, resulting in core damage. For all these sequences, a probability equal to 0.1 is used for failure of injection because of possible deformations in valves (this value is given by the OPRA, and was accepted in this review). It is important to point out that B&W analysis indicates that no deformation will occur at about 3900 psig. The following are the dominant accident sequences for this bin:

ATWS sequence number 9 = 2.1E-6/yr (OPRA = 3.1E-7/yr).
ATWS sequence number 12 = 5.2E-7/yr (OPRA = 6.2E-7/yr).
ATWS sequence number 21 = 2.1E-7/yr (OPRA = 3.1E-8/yr).
ATWS sequence number 67 = 1.9E-7/yr (OPRA = 3.8E-8/yr).
ATWS sequence number 27 = 1.4E-7/yr (OPRA = 1.4E-7/yr).
ATWS sequence number 73 = 1.3E-7/yr (OPRA = 2.1E-7/yr).

The main difference between this review and the OPRA comes from ATWS sequence number 9. This difference is caused by the probability used for failure of the function "primary relief." As discussed in Section B.1, the OPRA uses a value of 2.0E-2, while in this review a value of 0.17 is used. Note that, as discussed before, the value used in this review is subject to judgment and a sensitivity of total CD to this value will be provided in Section B.8

B.6 BIN VI ATWS SEQUENCES - BNL CD FREQUENCY: 3.4E-6/YR
OPRA CD FREQUENCY: 1.5E-6/YR

The sequences in this bin are very similar to the ones described in Section B.4 with the following differences: in this bin the injection of borated water is successful but the long-term cooling fails. The dominant accident sequences are:

ATWS sequence number 8 = 1.9E-6/yr (OPRA = 2.8E-7/yr).
ATWS sequence number 11 = 4.7E-7/yr (OPRA = 5.6E-7/yr).
ATWS sequence number 20 = 1.9E-7/yr (OPRA = 2.8E-8/yr).
ATWS sequence number 66 = 1.7E-7/yr (OPRA = 3.4E-8/yr).
ATWS sequence number 26 = 1.3E-7/yr (OPRA = 1.5E-7/yr).
ATWS sequence number 72 = 1.2E-7/yr (OPRA = 1.9E-7/yr).

The main difference between this review and the OPRA comes from sequence number 8 and, as in the previous section, it is caused by changes in the failure of the function "pressure relief."

B.7 SENSITIVITY ANALYSIS

As discussed in Section B.1 a sensitivity analysis to assess the effect of some parameters on ATWS CD frequency was performed, and the results are given below. The base case is that for which the CD frequency is given in the preceding sections; i.e., the total ATWS CD frequency is equal to 7.7E-6/yr for the base case.

B.7.1 Primary Relief Function

In the base case, it was assumed that Oconee-3 was operating with the PORV block valve closed for about 80% of the time the plant is in operation, and that for 20% of an equilibrium cycle the failure of the PORV to open may result in a peak pressure higher than 3900 psig. If this fraction of time is changed to 10% or 50%, the impact on the total ATWS core damage frequency (CD) is the following:

- Base Case (20%): CD = 7.7E-6/yr.

- 10%: CD = 5.4E-6/yr.

- 50%: CD = 1.5E-5/yr.

Note that, on the basis of the reactivity coefficients for an equilibrium cycle for Oconee-3, it is the judgment of the reviewer that the fraction of the time in which the peak pressure may be larger than 3900 psig will be smaller than that used in the base case (20%).

B.7.2 Failure to Inject Borated Water (RCS peak pressure <3500 psig)

In the OPRA, a value of 10^{-3} was used for the probability of failure to inject borated water for the cases in which the peak pressure will be smaller than 3500 psig. The same value used in the OPRA was used in this review for cases in which the MFW system remains on line, and a value of 2.0E-2 was used otherwise.

If for all the above cases a probability of failure equal to 5.0E-2 (a value used in the NRC ATWS task force analysis) is used, the total ATWS core damage frequency will change from 7.7E-6/yr to 1.5E-5/yr.

Note again that the values used in the base case are the more appropriate values in the opinion of the reviewer.

B.7.3 Failure to Inject Borated Water and Failure of Long-Term Cooling (RCS peak pressure >3900 psig).

In the OPRA, and in this review (base case), a value of 10^{-1} was used for the probability of failure to inject borated water, or the failure of long-term cooling for cases in which the peak pressure may exceed 3900 psig and a LOCA will result. These values were used because of the probability of deformation of valves in the injection paths. If these values are changed from 0.1 to 0.2, the total core damage frequency will change from 7.7E-6/yr to about 1.5E-5/yr.

B.8 SUMMARY

The dominant ATWS accident sequences were presented in previous sections, together with a comparison with the OPRA; in Table B.1 of the BNL review, ATWS core damage frequency for each of transient initiators and for each bin is given.

To summarize, the ATWS core damage frequency given in the OPRA is equal to 6.0E-6/yr; however, if this frequency is corrected to take into account an error in the probabilities given in sequences 14 and 15 event tree for ATWS turbine trip (OPRA, Appendix E, Figure E-1), the correct OPRA core damage frequency would be equal to about 3.6E-6/yr. This CD frequency is to be compared with the value of 7.7E-6/yr calculated in the base case of this review. Note that if the PORV block valve is open during all times the plant is in operation, as stated in a Nov 26, 1985 letter from H. B. Tucker (DPC) to H. R. Denton (NRC), the ATWS core damage frequency calculated in this review is equal to about 3.4E-6/yr.

A limited sensitivity analysis was also performed in this review (see Section B.7) and the resultant CD frequencies obtained vary from 5.4E-6/yr to 1.5E-5/yr.

To provide some perspective, the ATWS core damage frequencies obtained in PRAs performed for B&W plants are given below:

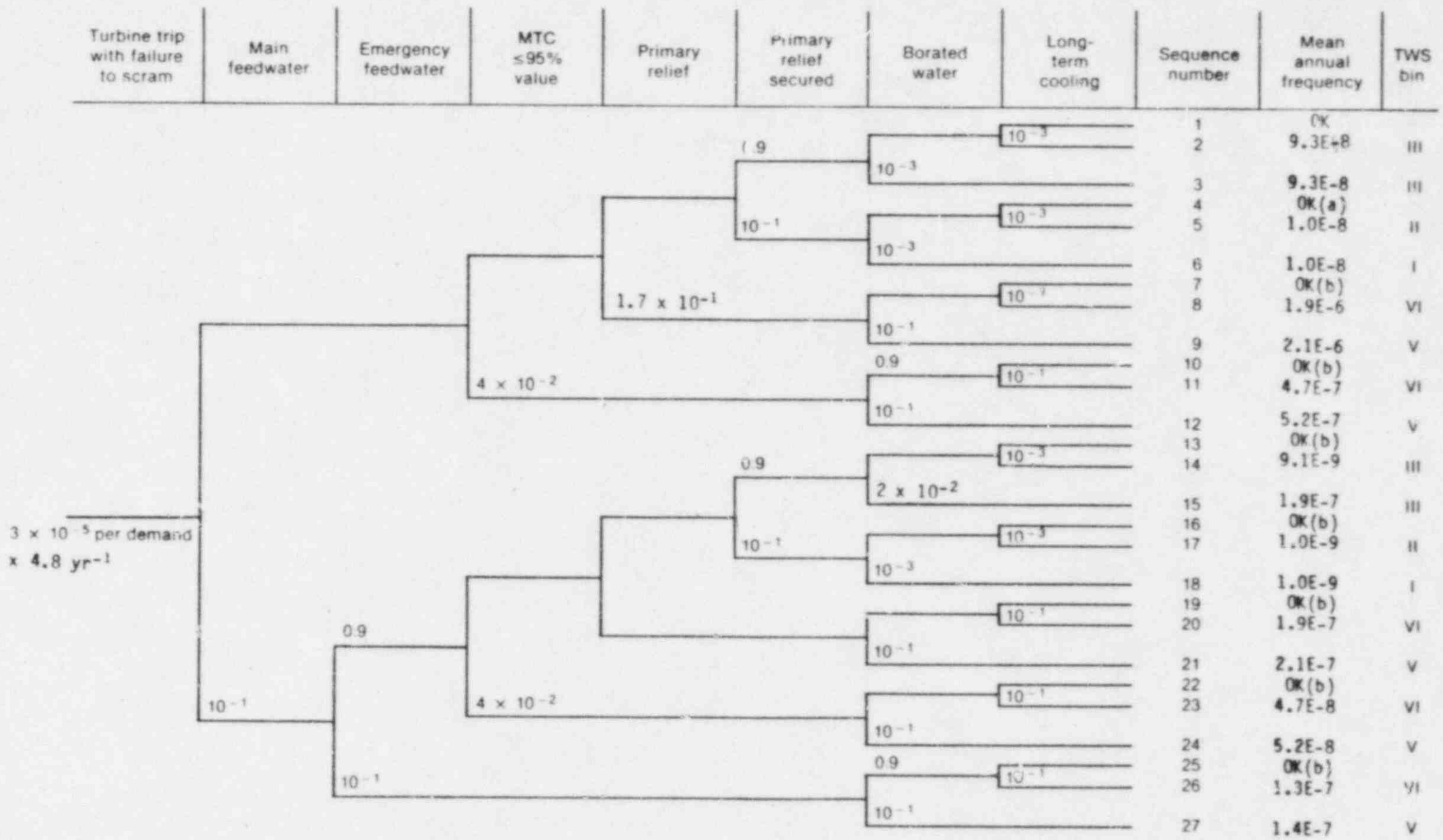
Crystal River - ~5.0E-7/yr (RPS failure = 1.5E-5/d).
ANO - ~2.8E-6/yr (RPS failure = 4.0E-6/d).
Oconee RSSMAP - ~8.0E-6/yr (RPS failure = 2.6E-5/yr).
Midland - very small.
NRC Task Force Analysis - 8.0E-5/yr.

Note that since not all the above PRAs present separate results for ATWS, the above CD frequencies are based on the dominant sequences. Note also that the assumptions used to obtain the ATWS core damage frequency are different in all the above PRAs.

Table B.1 ATWS Core Damage Frequency in the BNL Review - Base Case - Total CD Frequency: 7.7E-6/yr

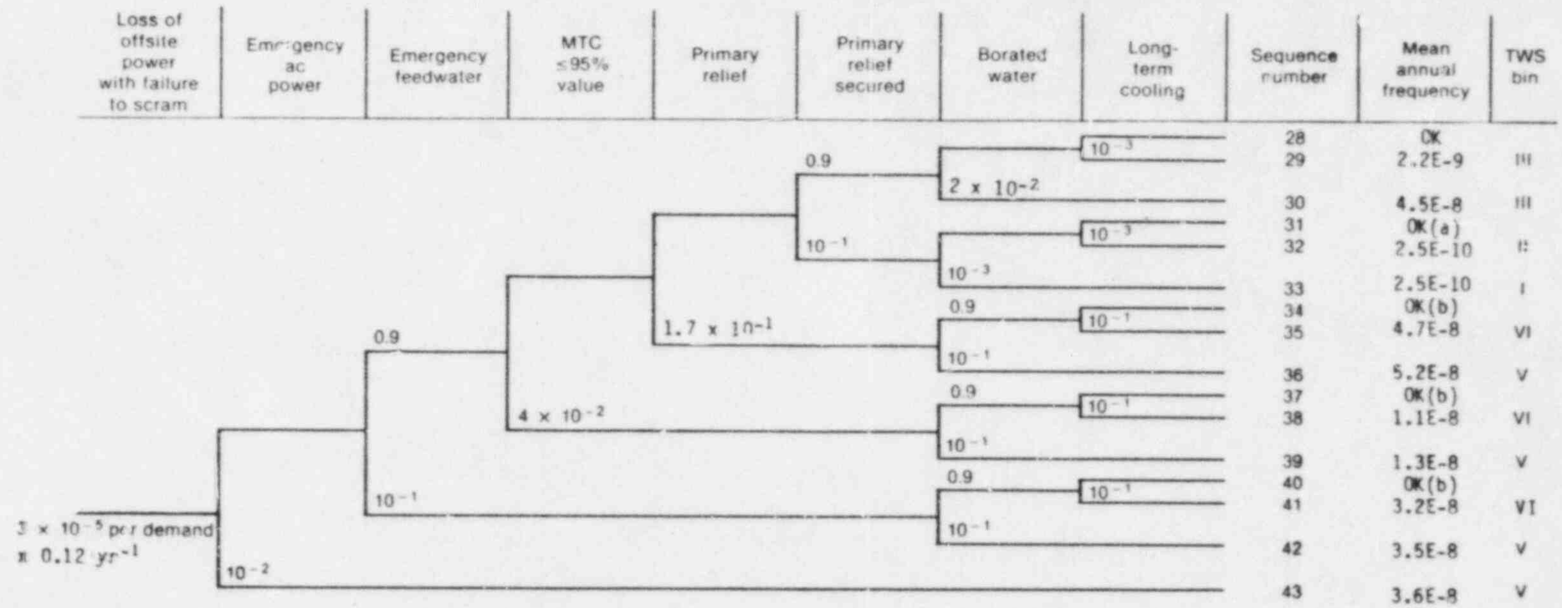
	T_T*	T_E*	T_{CV}*	T_F*
Bin I	1.1E-8	2.5E-10	3.6E-10	9.2E-10
Bin II	1.1E-8	2.5E-10	3.6E-10	9.2E-10
Bin III	3.9E-7	4.7E-8	6.9E-8	1.8E-7
Bin V	3.0E-6	1.4E-7	1.4E-7	3.7E-7
Bin VI	2.8E-6	9.0E-8	1.3E-7	3.4E-7
Total	6.2E-6	2.8E-7	3.4E-7	8.9E-7

*T_T - Turbine Trip.
T_E - LOOP.
T_{CV} - Loss of Condenser Vacuum.
T_F - Loss of Main Feedwater.



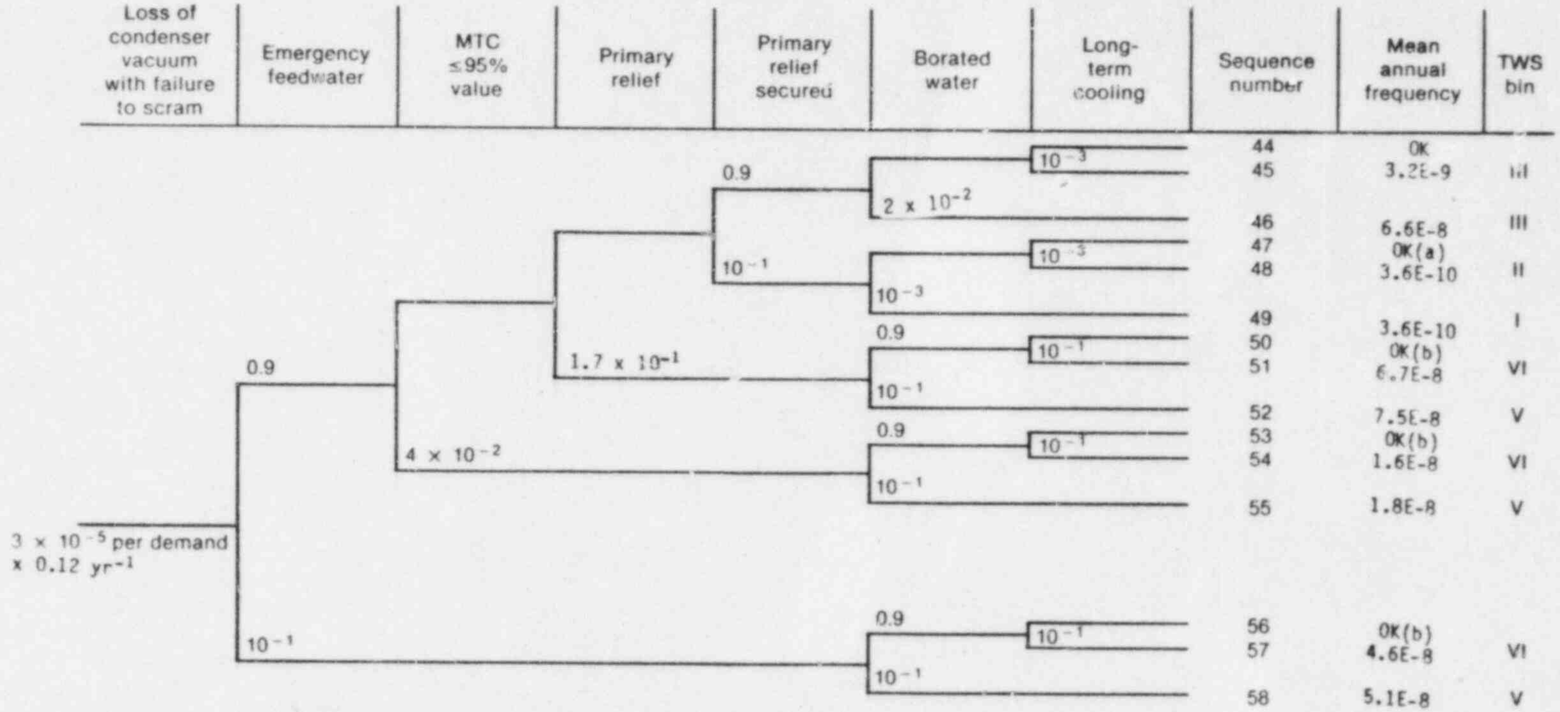
Note: OK = no core melt or LOCA
OK(a) = relief valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure B.1 Scoping event tree for turbine trip with failure to scram.



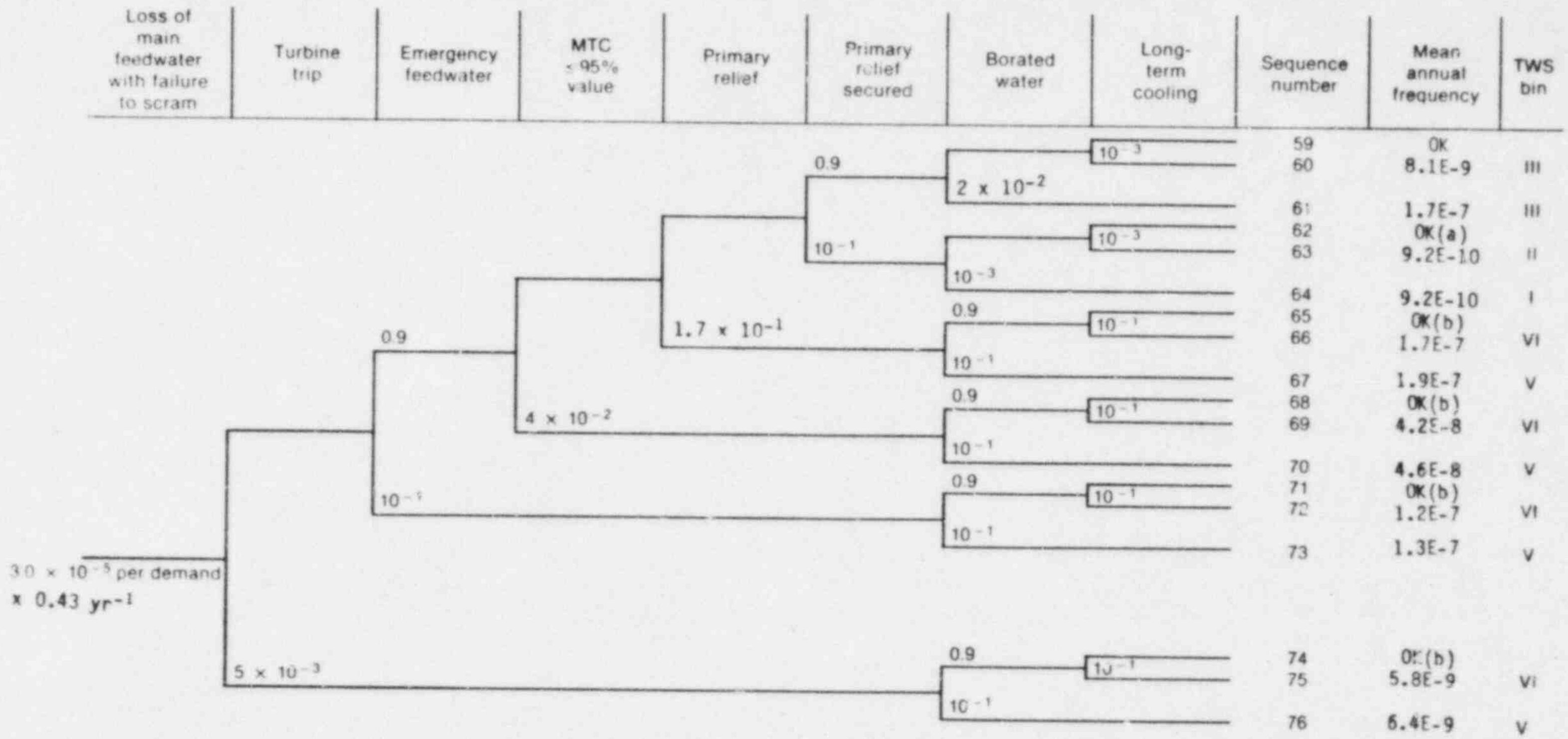
Note: OK = no core melt or LOCA
OK(a) = relief-valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure B.2 Scoping event tree for loss of offsite power with failure to scram.



Note: OK = no core melt or LOCA
OK(a) = relief-valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure B.3 Scoping event tree for loss of condenser vacuum with failure to scram.



Note: OK = no core melt or LOCA
OK(a) = relief-valve LOCA with successful mitigation
OK(b) = pressure-boundary LOCA with successful mitigation

Figure B.4 Scoping event tree for loss of main feedwater with failure to scram.

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A review of the Oconee-3 Probabilistic Risk Assessment (OPRA) was conducted with the broad objective of evaluating the contribution of the internally-generated accidents to the frequency of core damage. The review included a technical assessment of the assumptions and methods used in the OPRA study. The BNL staff reevaluated the main results of the study within the scope and general methodological framework, including both qualitative and quantitative analyses of accident initiators, and accident sequences which result in core damage. The effect of uncertainties was considered throughout the review process, and the uncertainty bands for the core damage frequency were quantified.

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