Browns Ferry Nuclear Plant Unit 2

Probabilistic Risk Assessment Individual Plant Examination

Revision 1A

Consisting of:

Interim Order No. 1 – April 13, 1995
Interim Order No. 2 – September 12, 1995
Interim Order No. 3 – October 23, 1997

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APR 1 3 1995

Probabilistic Risk Assessment (PRA) Manual Holders

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 2 PRA INDIVIDUAL PLANT EXAMINATION REVISION 1 - INTERIM ORDER (IO) NO. 1

The purpose of this IO is to correct inconsistencies identified subsequent to issuance of the Browns Ferry Unit 2 Individual Plant Examination Revision 1.

This IO shall be effective upon approval and shall be incorporated into the BFN Unit 2 Individual Plant Examination during the next revision.

Henry L. Jones
Senior Engineering Specialist
EDB 1A-BFN

for H. L. Williams, Manager
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cc (Attachments):
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PLDNE103/192/15

TVA

REVISION LOG
IO #1
REVISION 1

Title: PROBABILISTIC SAFETY ASSESSMENT - REPORT

Revision No.	DESCRIPTION OF REVISION	Date Approved
0	Initial Issue	9/1/92
1	Revised to incorporate the following items: 1. Addresses comments identified prior to issue of Revision 0, but were not incorporated into Revision 0. 2. Incorporates design changes physically made to the plant between December 1, 1991 and May 31, 1993. 3. Updates the initiating event frequencies and diesel generator data using plant-specific data.	8/12/94
	R92 950413 800	
1	INTERIM ORDER No. 1 Rims ______ Revised To ensure Consistency with The BFN Multi-Unit PRA.	APR 13 1995

PLDNE103/192/13

TVA 10534 (EN DES-4-78)

1. EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This report documents the work performed by Tennessee Valley Authority (TVA) in accordance with the U.S. Nuclear Regulatory Commission (NRC) Generic Letter No. 88-20 (Reference 1-1), which requested each utility to perform an individual plant examination (IPE) "(1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant, (3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents." To satisfy these requirements, a Level 1 and Level 2 probabilistic safety assessment (PSA) was performed for Unit 2 of the Browns Ferry Nuclear Plant. The PSA, Revision 0, was performed by an integrated team of engineers and PSA specialists from TVA and PLG, Inc., with support from ERIN Engineering, Inc., Gabor, Kenton and Associates; and EQE Engineering, Inc. Revision 1 Interim Order #1 was performed by an integrated team of engineers and PSA specialists from TVA and PLG, Inc. The results are applicable to Unit 2 only.

TVA's original overall objectives of the Revision 0 PSA program were to

- Meet the NRC requirements for IPEs as set forth in Generic Letter No. 88-20 (Reference 1-1) and in NUREG-1335 (Reference 1-2).
- Develop a plant-specific Level 2 PSA model for Browns Ferry Unit 2 based on the plant design as of December 1991.
- Develop and apply databases for common cause failure parameters and human error rates. Generic databases were used for initiating event frequencies, component failure rates, and maintenance unavailabilities.
- Develop point estimate and uncertainty distribution results for the frequency of core damage and a full spectrum of radioactive release categories for Browns Ferry Unit 2.
- Determine the underlying risk controlling factors and key sources of uncertainty in developing the risk estimates.
- Identify and implement opportunities for safety enhancement.

TVA's overall objectives of the Revision 1 PSA were to:

- Address comments identified prior to issue of revision 0 but were not incorporated into revision 0.
- Incorporates design changes physically made to the plant between December 1, 1991, and May 31, 1993.

- Updates the initiating event frequencies and diesel generator data using plant-specific data.

The Level 2 evaluation was not revised as part of the Revision #1 Interim Order 1PSA effort. The Revision 1 Interim Order #1PSA provides a more realistic and representative CDF. The relative rankings of the accident sequences leading to core damage has changed, but at the same time the absolute plant risk has decreased since the probability of having an accident scenario leading to core damage has decreased by a factor of approximately 6. Therefore, from a Level 2 standpoint, the binning of the plant damage states may have changed showing a different distribution of accident scenarios. However, the overall risk of having a core damage scenario which may lead to a potential release from the containment has been reduced significantly. It is therefore concluded that the Level 2 analysis performed for the Revision 0 PSA is conservative with respect to the Revision 1 PSA Level 1 results.

The purpose of this summary is to present the results of the PSA on Browns Ferry Unit 2. These results include an estimate of the total core damage frequency (CDF); uncertainties in the estimated CDF; and the key plant damage states and release categories. This summary also provides the sequences, systems, and sources of uncertainty that are the significant contributors to the results. In addition, information is provided on the nature, timing, and magnitude of potential releases of radioactive material.

1.2 PLANT FAMILIARIZATION

The Browns Ferry Nuclear Plant is located on the north shore of Wheeler Lake at Tennessee River mile 294 in Limestone County, Alabama. The site is approximately 10 miles southwest of Athens, Alabama, and 10 miles northwest of Decatur, Alabama. The plant consists of three units, each with a rated power level of 3,293 MWt. Unit 2 is the only unit currently operating.

Unit 2 is a single-cycle forced-recirculation boiling water reactor (BWR) nuclear steam supply system supplied by General Electric Corporation. Major structures at Browns Ferry Unit 2 include a reactor building with three Mark I drywell containments, a turbine building, a control bay, and an intake pumping station.

A detailed description of the plant site, facilities, and safety criteria is documented in the Browns Ferry Final Safety Analysis Report (Reference 1-3).

1.3 OVERALL METHODOLOGY

The Browns Ferry Unit 2 PSA is founded on a scenario-based definition of risk (Reference 1-4). In this application, "risk" is defined as the answers to three basic questions:

1. What can go wrong?
2. What is the likelihood?
3. What are the consequences?

Question 1 is answered with a structured set of scenarios that is systematically developed to account for design and operating features specific to Browns Ferry Unit 2. Question 2 is answered with a prediction or estimate of the frequency of occurrence of each scenario

identified in the answer to question 1. Since there is uncertainty in that frequency, the full picture of likelihood is conveyed by a probability curve—a curve that conveys the state of knowledge, or confidence, about that frequency.

Question 3 is answered in a Level 2 PSA in terms of the key characteristics of radioactive material releases that could result from the scenarios identified. The results reported here are based on the methods that conform to the NRC guidelines (Reference 1-1, Appendix 1) and the IEEE/ANS "PSA Procedures Guide" (Reference 1-5).

A large fraction of the effort needed to complete this PSA was to develop a plant-specific model to define a set of accident sequences. This model contains a large number of scenarios that have been systematically developed from the point of initiation to termination. A series of event trees is used to systematically identify the scenarios. Given the knowledge of the event tree structures, accident sequences are identified by specifying:

1. The initiating event.
2. The plant response in terms of combinations of systems and operator responses.
3. The end state of the accident sequence.

The RISKMAN® PC-based software system (Reference 1-6) was used to construct effectively a single, large tree for Level 1 and a second tree for Level 2. The sequences in the Level 1 analysis start with an initiating event and terminate in plant damage states. Plant damage states are the starting points for the Level 2 event trees. Level 2 event trees are quantified separate from the Level 1 trees. Release categories are used as the end states of the Level 2 event trees.

The initiating events and the event tree branching frequencies are quantified using different types of models and data. The system failures that contribute to these events are analyzed with the use of fault trees that relate the initiating events and event tree branching frequencies to their underlying causes. These causes are quantified, in turn, by application of models and data on the respective unavailabilities due to hardware failure, common cause failure, human error, and test and maintenance unavailabilities. The frequencies of initiating events, the hardware failure rates of the components, and operator errors were obtained using either generic data or a combination of generic and plant-specific data.

Dependency matrices are developed from a detailed examination of the plant systems to account for important interdependencies and interactions that are highly plant specific. To facilitate a clear definition of plant conditions in the scenarios, separate stages of event trees are provided for the response of the support systems (e.g., electric power and cooling water), the frontline systems [e.g., high pressure coolant injection (HPCI) and residual heat removal (RHR)], and the containment phenomena; e.g., containment overpressurization failure. The containment event tree is used in the Level 2 PSA. A detailed definition of plant damage states provides the interface between the Level 1 and Level 2 event trees.

The systematic, structured approach that is followed in constructing the accident scenario model provides assurance that plant-specific features are identified. It also provides insights into the key risk controlling factors.

1.4 SUMMARY OF MAJOR FINDINGS

The major findings of the Browns Ferry Unit 2 Level 2 PSA are presented in this section. The results delineate the principal contributors to risk, and provide insights into plant and operational features relevant to safety.

The Revision 0 PSA included both the Level 1 and Level 2 results. However, for the Revision 1 PSA and Interim Order #1, only the Level 1 portion of the plant model was evaluated and revised. The Level 2 portion of the plant model was not revised.

1.4.1 TOTAL CORE DAMAGE FREQUENCY

The total CDF for Browns Ferry Unit 2 was found to be 7.6x 10^{-6} per reactor-year.* For this analysis, core damage is assumed for any sequence in which sustained core uncovery occurs. The results for CDF were developed in terms of a mean point estimate, as required in NUREG-1335 (Reference 1-2). Presentation of the total CDF is in terms of the uncertainty distribution shown in Figure 1-1.

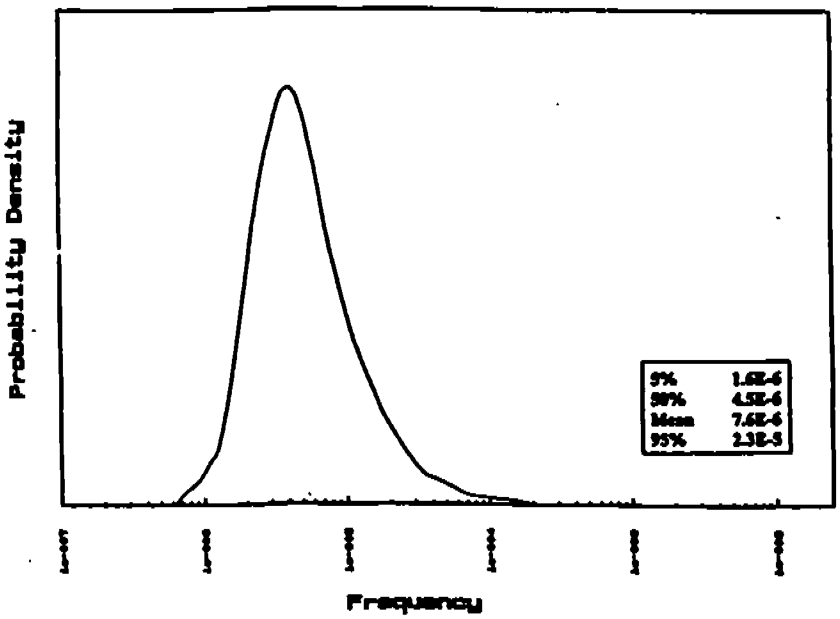


Figure 1-1. Uncertainty Distribution for Browns Ferry Unit 2 Core Damage Frequency

A comparison of this study with other PSAs on other plants that used similar methods, databases, and work scopes is given in the following table. The calculated mean CDF for Browns Ferry Unit 2 is of the same order of magnitude as Peach Bottom Unit 2 and Grand Gulf Unit 1, and an order of magnitude lower than that reported for Shoreham and Nine Mile Point Unit 2.

*The unit for the core damage frequency is events per nuclear-powered electric generating unit per calendar year. This definition is abbreviated to "per reactor-year."

Plant	Flood Included	Mean CDF (per reactor-year)	Reference
Shoreham	Yes	5.5×10^{-5}	1-7
Nine Mile Point Unit 2	Yes	3.0×10^{-5}	1-8
Browns Ferry Unit 2	Yes	7.6×10^{-6}	This Study
Peach Bottom Unit 2	No	4.5×10^{-6}	1-9
Grand Gulf Unit 1	No	4.0×10^{-6}	1-10

Factors that contribute to the results for Browns Ferry Unit 2 are summarized below:

- The accident sequences that were analyzed are those initiated by internal events and internal floods, in accordance with IPE requirements. Sequences initiated by internal fires, seismic events, and other external events have not been considered.

- The current results do not reflect any future plant or procedural changes that TVA may decide to make to improve safety.

- The Revision 0 results were obtained using industry data for failure rates, generic data for maintenance unavailabilities, and initiating event frequencies (Reference 1-11). The Revision 1 effort included using plant specific data to update failure rates for selected components and initiating events frequencies. The common cause parameters of the multiple Greek letter model were used in this study and were estimated with the benefit of a plant-specific screening of industry common cause event data in accordance with NUREG/CR-4780 (Reference 1-12).

- Operator actions to recover from adverse plant states were considered up to the point of core damage; however, further accident management actions that would be expected after the onset of core damage were not considered. It is important to note that in this study, operator actions considered to prevent core damage were limited.

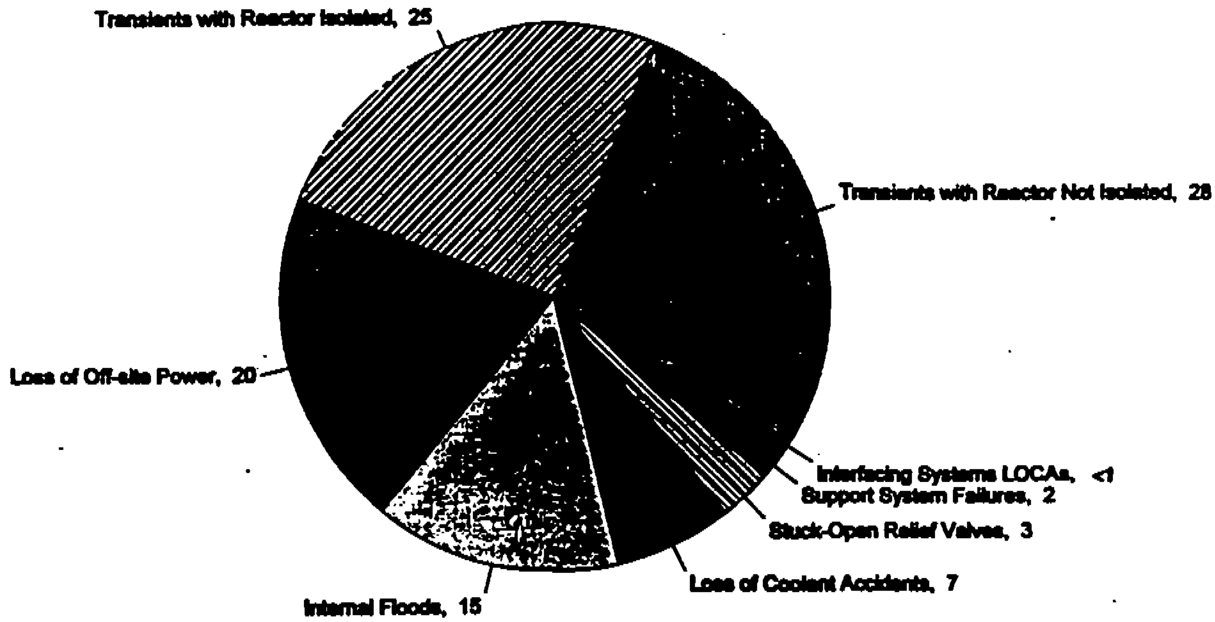
- Scenarios initiated by internal floods at Browns Ferry Unit 2 contribute approximately 15% to the total CDF.

1.4.2 CONTRIBUTORS TO TOTAL CORE DAMAGE FREQUENCY

In the quantification of the Level 1 event sequence models, the principal contributors to the CDF were identified from several vantage points. The results and contributors are summarized in this section. Section 3.4 describes the results in detail, along with the accident sequence screening process. Causes for individual system failures are listed in each systems analysis notebook.

1.4.2.1 Important Core Damage Sequence Groups

The importance of initiating events was examined by determining the contributions of core damage sequences grouped by initiating event. The ranked results are shown in Figure 1-2 and Table 1-1 for major initiating event categories.



TOTAL CDF = 7.6×10^{-6} PER REACTOR-YEAR

Figure 1-2. Browns Ferry Unit 2 Core Damage Frequency by Initiating Event Category

Table 1-1. Initiating Event Group Contributions to Core Damage Frequency

Initiating Event Category	Mean CDF (per reactor-year)	Percentage of Total
Transients with Reactor Not Isolated	2.14×10^{-6}	28
Transients with Reactor Isolated	1.91×10^{-6}	25
Loss of Offsite Power	1.48×10^{-6}	20
Internal Floods	1.14×10^{-6}	15
Loss of Coolant Accidents	5.14×10^{-7}	7
Stuck-Open Relief Valves	1.86×10^{-7}	3
Support System Failures	1.70×10^{-7}	2
Interfacing System LOCAs	4.63×10^{-8}	<1
Total	**7.6×10^{-6}**	**100**

The highest single initiating event is a loss of offsite power (LOSP). Table 1-1 does not reveal this since it presents some categories of multiple initiation. The LOSP initiators include station blackout sequences (failure of all diesel generators) and nonstation blackout scenarios in which core damage resulted from other failures. These other failures include battery board failures (resulting in loss of high pressure injection and failure to achieve low pressureinjection) and decay heat removal failures. Overall, the LOSP initiated sequences account for 20% of CDF.

Transients with the reactor not isolated contribute 28% to the core damage frequency. Turbine trip and loss of feedwater are two specific examples of initiators in this group.

Transients with the reactor isolated as a result of the initiator contribute 25% to the core damage frequency. Closure of the main steam isolation valves (MSIV) and turbine trip without bypass are two specific examples of initiators in this group.

Scenarios initiated by internal floods contribute 15% to the core damage frequency. No internal flooding scenarios lead directly to core damage but require additional hardware failures. Flooding initiators were postulated in the Unit 2 reactor building, in the Unit 1 or Unit 3 reactor building, in the turbine building, and at the intake pumping station.

Large and medium LOCAs and interfacing systems LOCAs (i.e., when the boundary between a high and a low pressure system fails and the lower pressure system overpressurizes) make up 7% of the total CDF.

Scenarios initiated by the inadvertent opening of one or more relief valves contribute 2% to the core damage frequency. Three distinct initiators are considered: opening of one safety relief valve (SRV), opening of two SRVs, and opening of three or more SRVs.

Support system failure initiators (specifically, loss of plant air, loss of raw cooling water, loss of unit preferred power, loss of either I&C bus 2A or 2B, or instrument tap failures) also contribute 2% to the total CDF.

The preceding paragraphs considered the contribution to the total CDF from groups of initiating events. The sequences leading to core damage were also reviewed to identify common functional failures. Approximately ~~88%~~ of the total CDF is associated with the three accident sequence groups shown in Table 1-2. 61%

Table 1-2. Functional Failure Group Contributions to Core Damage Frequency

Accident Sequence Group	Mean CDF (per reactor-year)	Percentage of Total
Transient with Vessel at High Pressure	5.2×10^{-7}	7%
Transient Followed by Loss of Vital DC Power (250V boards 2 and 3)	2.5×10^{-6}	33%
Anticipated Transient without Scram (ATWS)	1.6×10^{-6}	21%

Events in which the reactor remains at high pressure contribute 37% to the total CDF. These scenarios are distributed among the transient initiators and are dominated by the reactor not being depressurized to allow low pressure makeup on loss of HPCI, RCIC and control rod drive (CRD). The Level 1 evaluation that resulted in identifying the sequences contributing 37% to the CDF was based on a conservative depressurization criterion (vessel depressurization prior to vessel level reaching -190 inches). As described in Section 4, the Level 2 analysis considered additional credit for late depressurization.

Sequences in which 250V DC battery boards 2 and 3 both fail make up 32% of the total CDF. These sequences require depressurization of the reactor vessel and use of the low pressure injection systems, to prevent core damage; however, failure of 250V DC battery boards 2 and 3 impact the availability of low vessel pressure permissive signals for low pressure coolant injection (LPCI) and core spray.

Sequences without a reactor scram (ATWS) contribute approximately 21% to the total CDF. Such sequences may lead to core damage if alternate means of controlling reactivity (for example, boron injection) fail.

Table 1-3 summarizes the contribution of the top sequence to the CDF at Browns Ferry Unit 2. This sequence accounts for 3.6% of the CDF. Other sequences involving loss of high pressure injection are similar.

Table 1-3. Analysis of the Top Sequence

Sequence Element	Element Description	Mean Frequency (per reactor-year)
Initiating Event	Flood in the Turbine Building	2.23×10^{-2}
Additional Equipment or Operator Failures after Initiating Event	Failure of 250V DC Battery Boards 2 & 3	2.5×10^{-5}
Recovery Actions	Board Failure not Recoverable or Attempts with Alternate Charger Fail	5.30×10^{-1}
Consequential Effects of Above Failures on Plant Performance	High Pressure Injection Unavailable. Low Vessel Pressure Permissive Signals Unavailable Causing Failure of LPCI and Core Spray	1.0
Remaining Plant Response	Successful operation of remaining systems and actions.	6.08×10^{-1}
Total Sequence Frequency (product of contributors)		2.74×10^{-7}

1.4.2.2 Analysis of Individual Sequences

No single core damage sequence was found to dominate the total frequency of core damage. A large number of sequences make up the total CDF. Table 1-4 provides information on the distribution of core damage sequences across the frequency range.

Table 1-4. Breakdown of Core Damage Sequences in Each Frequency Range

Frequency Range (events per year)	Number of Sequences	Percentage of CDF
10^{-7} to 10^{-6}	6	13.6
10^{-8} to 10^{-7}	110	35.2
10^{-9} to 10^{-8}	1,072	37.8
$<10^{-9}$	3,553	14.2

The following presents a brief description of the 10 highest ranking sequences to the CDF. Six of the top ten sequences involve failures of Battery Boards 2 & 3. Two sequences involve failure of the Reactor Protection System resulting in an ATWS. The remaining sequences are a loss of offsite power which progresses to a station blackout and a flood in which high pressure injection failed and the vessel was not depressurized.

- Turbine Building Flood with failure of Battery Boards 2 and 3 and no recovery using the spare charger (3.6% CDF)

- Closure of all MSIVs with failure of Battery Boards 2 and 3 and no recovery using the spare charger (3.0% CDF)

- Total Loss of feedwater with failure of Battery Boards 2 and 3 and no recovery using the spare charger (2.0% CDF)

- Loss of Condenser Vacuum with failure of Battery Boards 2 and 3 and no recovery using the spare charger (1.8% CDF)

  Turbine Trip without Bypass with failure of Battery Boards 2 and 3 and no recovery using the spare charger (1.6% CDF).

- Turbine Building Flood with HPCI and RCIC unavailable and the vessel remains at high pressure (1.6% CDF).

- A Loss of offsite power in which emergency diesel generators were successful but with failure of Battery Boards 2 and 3 and no recovery using the spare charger (1.2% CDF).

- Turbine Trip with failure to scram (ATWS) and failure of the standby liquid control system (1.1% CDF).

- A Loss of offsite power with failure of the station diesel generators (four Unit 1/2 diesel generators and four Unit 3 diesel generators with no power recovery in six hours (1.0% CDF).

- Turbine Trip with failure to scram (ATWS) and failure to initiate the standby liquid control system (1.0% CDF).

Section 3.4 contains a detailed discussion of the top 10 sequences as well as a listing of the top 100 core damage sequences.

1.4.2.3 Important Operator Actions

The importance of a specific operator action was determined by summing the frequencies of the sequences involving failure of that action and comparing that sum to the total CDF. The importance is the ratio of that sum to the total CDF. The importance analysis that was performed for Browns Ferry Unit 2 is discussed more fully in Section 3.4.

Table 1-5 summarizes the important operator action failures ranked in order of their impact on the total CDF. The operator actions to recover electric power are not included in Table 1-5 because they are a complex function of the time available and the specific equipment failures involved.

Table 1-5. Browns Ferry Unit 2 Important Operator Actions

	Operator Action	Importance
1.	Manual Depressurization of the Reactor Vessel using the Safety Relief Valves	.060
2.	Initiate Standby Liquid Control System, Given an ATWS with an Unisolated Vessel	.034
3.	Align RHR for Suppression Pool Cooling	.027
4.	Initiate Standby Liquid Control System, Given an ATWS with Vessel Isolated	.019
5.	Align Alternate Injection to Reactor Vessel Via the Unit 1 to Unit 2 RHR Crosstie	.015
6.	Control Reactor Vessel Level at Low Pressure Using RHR or Core Spray	.011
7.	Align and Start a RHRSW Swing Pump After a LOSP With Degraded EECW	.008
8.	Inhibit Automatic Depressurization During an ATWS	.007
9.	Start RHR or Core Spray Pumps, Given High Pressure Injection (HPCI, RCIC) Has Failed	.007
10.	Transfer Unit 1 and Unit 2 4-kV Unit Boards to 161-kV Power Given Loss of 500-kV Power	.003

1.4.2.4 Important Plant Hardware Characteristics

An importance analysis of plant system failure modes to the total CDF was also performed. Only hardware failures involving the system itself are considered in Table 1-6, which provides a ranking in order of their impact on the total CDF.

Table 1-6. Browns Ferry Unit 2 Important Systems

System	% CDF
250V Battery Boards	51
Reactor Protection System	20
Residual Heat Removal System	20
Diesel Generators	15
Residual Heat Removal Service Water System	9
Main Steam System including Turbine Trip	8
High Pressure Coolant Injection System	7
Shared Actuation Instrumentation	7
Reactor Core Isolation Cooling System	6
Control Rod Drive Hydraulic System	4
Standby Liquid Control System	4
Condensate and Feedwater System	1

The system importance means the fraction of the CDF involving partial or complete failure of the indicated system. These importance measures are not strictly additive because multiple system failures may occur in the same sequence. The importance rankings account for failures within the systems that lead to a plant trip, or failures that limit the capability of the plant to mitigate the cause of a plant trip. Consequential failures resulting from dependencies on other plant systems [e.g., the loss of drywell control air due to failure of reactor building closed cooling water (RBCCW)] are not included in this importance ranking.

A further discussion of system importance and the results of additional sensitivity analysis are provided in Section 3.4.

Note: The following sections (1.4.2.5, 1.4.3, 1.4.3.1, 1.4.3.2) were not updated since the Level 2 analysis was not revised as part of the Revision 1, Interim Order #1 PSA effort.

1.4.2.5 Plant Damage States

One output from the Level 1 event sequence model is a description of the plant damage state (PDS) at the time of core damage. This plant damage state strongly influences the performance of the containment and the magnitude of fission product release that is assessed in the Level 2 analysis, as described in Section 4.

Table 1-7 provides the frequencies of different plant damage states associated with core damage. The results in Table 1-7 only account for the impact on containment integrity of the accident sequence up to the time of core damage. In the context of the PSA, the terms "early failure" and "late failure" refer to specific transient characteristics. For an "early failure" to occur, the transient must involve phenomena that could cause the drywell

pressure to rise rapidly and exceed the ultimate containment pressure. An unmitigated ATWS is an example of this type of transient. "Late failures" are those in which reactor vessel injection is accomplished but the containment heat removal is not accomplished. The result is a slow rise in containment pressure to the failure point before core damage occurs. As examples, the Level 1 analysis considers failures to isolate containment penetrations, containment bypass from line breaks outside containment, and preexisting leaks. Table 1-7 results do not account for challenges to containment from severe accident phenomena such as hydrogen burns and concrete degradation, which are analyzed in Level 2.

As shown in Table 1-7, over 95% of the core damage events would be associated with an intact containment, and 1.8%, 0.8%, and 0.4% would be associated with late containment failure, early containment failure, and containment bypass, respectively. Core damage sequences accompanied by a containment bypass lead to a greater fission product release.

Table 1-7. Core Damage Frequency Breakdown for Browns Ferry Unit 2 by Major PDS Group

Containment State	Frequency (per reactor-year)	CDF (percent)
Intact	4.6×10^{-5}	97.0
Late Failure	8.7×10^{-7}	1.8
Early Failure	4.0×10^{-7}	0.8
Bypass	2.0×10^{-7}	0.4
Total	4.8×10^{-5}	100.0

1.4.3 RESULTS FOR RELEASE FREQUENCY

This section summarizes the results for the Level 2 analysis which established the release category groups and estimated the frequency of the release categories. Sequences quantified by the Level 1 analysis (representing the sequences leading to core damage), are further processed through the containment event tree (CET). The CET top events represent the systemic and phenomenological events in the accident progression from core uncovery to possible containment failure and release to the environment. The results of the CET quantification process are summarized in Table 1-8. The same results are also displayed in Figure 1-3.

Table 1-8. Summary of Release Frequency Results by Release Category Groups

Release Category Groups	Description	Percentage of CDF
I	Large, Early Containment Failures and Large Bypasses	46
II	Small, Early Containment Failures and Small Bypasses	0

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Probabilistic Risk Assessment (PRA) Manual Holders

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 2 PRA INDIVIDUAL PLANT EXAMINATION REVISION 1 - INTERIM ORDER (IO) NO. 2

950920T0389

The purpose of this IO is to correct inconsistencies identified subsequent to issuance of the Browns Ferry Unit 2 Individual Plant Examination Revision 1, Interim Order 1.

This IO shall be effective upon approval and shall be incorporated into the BFN Unit 2 Individual Plant Examination during the next revision.

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Title: PROBABILISTIC SAFETY ASSESSMENT - REPORT

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REVISION LOG
IO #1
REVISION 1

Revision No.	DESCRIPTION OF REVISION	Date Approved
0	Initial Issue	9/1/92
1	Revised to incorporate the following items: 1. Addresses comments identified prior to issue of Revision 0, but were not incorporated into Revision 0. 2. Incorporates design changes physically made to the plant between December 1, 1991 and May 31, 1993. 3. Updates the initiating event frequencies and diesel generator data using plant-specific data.	8/12/94
	R92 950413 800	
1	INTERim ORDER No. 1 Rims ______ Revised To ensure Consistency with The BFN multi-unit PRA	APR 13 1995
	R92 950912 800	
2	INTERIM ORDER No. 2 RIMS: ~~R06950911002~~ This interim order revises THE METHODOLOGY FOR Calculating Human action importances WHICH ARE subsumed in SPLIT FRACTIONS containing OTHER BASIC EVENTS.	9/12/95

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Table 1-4. Breakdown of Core Damage Sequences in Each Frequency Range		
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- Closure of all MSIVs with failure of Battery Boards 2 and 3 and no recovery using the spare charger (3.0% CDF)
- Total Loss of feedwater with failure of Battery Boards 2 and 3 and no recovery using the spare charger (2.0% CDF)
- Loss of Condenser Vacuum with failure of Battery Boards 2 and 3 and no recovery using the spare charger (1.8% CDF)

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- A Loss of offsite power with failure of the station diesel generators (four Unit 1/2 diesel generators and four Unit 3 diesel generators with no power recovery in six hours (1.0% CDF).

- Turbine Trip with failure to scram (ATWS) and failure to initiate the standby liquid control system (1.0% CDF).

Section 3 4 contains a detailed discussion of the top 10 sequences as well as a listing of the top 100 core damage sequences.

1.4.2.3 Important Operator Actions

The importance of a specific operator action was determined by summing the frequencies of the sequences involving failure of that action and comparing that sum to the total CDF. The importance is the ratio of that sum to the total CDF. For split fractions containing basic events other than human actions, a weighting factor based on the contribution of the human action to the split faction was used. The importance analysis that was performed for Browns Ferry Unit 2 is discussed more fully in Section 3.4.

Table 1-5 summarizes the important operator action failures ranked in order of their impact on the total CDF. The operator actions to recover electric power are not included in Table 1-5 because they are a complex function of the time available and the specific equipment failures involved.

Table 1-5. Browns Ferry Unit 2 Important Operator Actions

	Operator Action	Importance
1.	Manual Depressurization of the Reactor Vessel using the Safety Relief Valves	.060
2.	Align RHR for Suppression Pool Cooling	.059
3.	Initiate Standby Liquid Control System, Given an ATWS with an Unisolated Vessel	.035
4.	Cross-tie Unit 1 RHR Pumps and Heat Exchanger to Unit 2 Torus	.026
5.	Start RHR/CS pumps for LPCI, L1 signal not anticipated	.022
6.	Initiate Standby Liquid Control System, Given an ATWS with Vessel Isolated	.020
7..	Control Reactor Vessel Level at Low Pressure Using RHR or Core Spray	.011
8.	Align and Start a RHRSW Swing Pump After a LOSP With Degraded EECW	.008
9.	Inhibit Automatic Depressurization During an ATWS	.007
10.	Depressurize with the Turbine Bypass Valves after Loss of HPCI and RCIC	.006

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Probabilistic Risk Assessment (PRA) Manual Holders

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 2 PRA INDIVIDUAL PLANT EXAMINATION REVISION 1 - INTERIM ORDER (IO) NO. 3

The purpose of this IO is to correct inconsistencies identified by Problem Evaluation Report BFPER970754 for the BFN Unit 2 Individual Plant Examination.

This IO shall be effective upon approval and shall be incorporated into the BFN Unit 2 Individual Plant Examination during the next revision.

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Senior Engineering Specialist
PEC 2A-BFN

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HLJ:DS
Attachments
cc (Attachments):
RIMS, WT 3B-K

TVA

Title: PROBABILISTIC SAFETY ASSESSMENT - REPORT	REVISION LOG REVISION 1, IO # 1

Revision No.	DESCRIPTION OF REVISION	Date Approved
0	Initial Issue	9/1/92
1	Revised to incorporate the following items: 1. Addresses comments identified prior to issue of Revision 0, but were not incorporated into Revision 0. 2. Incorporate design changes physically made to the plant between December 1, 1991 and May 31, 1993. 3. Updates the initialing event frequencies and diesel generator data using plant-specific data.	8/12/94
1	Interim Order No. 1, RIMS R92 950413 800 Revised to ensure consistency with the BFN Multi-Unit PRA	4/13/95
2	Interim Order No. 2, RIMS R92 950912 800 This interim order revises the methodology for calculating human action importances which are subsumed in split fractions containing other basic events.	
3	Interim Order No. 3 This Interim Order corrects inconsistencies identified by PER BFPER970754.	

Interim Order #1

identified in the answer to question 1. Since there is uncertainty in that frequency, the full picture of likelihood is conveyed by a probability curve—a curve that conveys the state of knowledge, or confidence, about that frequency.

Question 3 is answered in a Level 2 PSA in terms of the key characteristics of radioactive material releases that could result from the scenarios identified. The results reported here are based on the methods that conform to the NRC guidelines (Reference 1-1, Appendix 1) and the IEEE/ANS "PSA Procedures Guide" (Reference 1-5).

A large fraction of the effort needed to complete this PSA was to develop a plant-specific model to define a set of accident sequences. This model contains a large number of scenarios that have been systematically developed from the point of initiation to termination. A series of event trees is used to systematically identify the scenarios. Given the knowledge of the event tree structures, accident sequences are identified by specifying:

1. The initiating event.
2. The plant response in terms of combinations of systems and operator responses.
3. The end state of the accident sequence.

The RISKMAN® PC-based software system (Reference 1-6) was used to construct effectively a single, large tree for Level 1 and a second tree for Level 2. The sequences in the Level 1 analysis start with an initiating event and terminate in plant damage states. Plant damage states are the starting points for the Level 2 event trees. Level 2 event trees are quantified separate from the Level 1 trees. Release categories are used as the end states of the Level 2 event trees.

The initiating events and the event tree branching frequencies are quantified using different types of models and data. The system failures that contribute to these events are analyzed with the use of fault trees that relate the initiating events and event tree branching frequencies to their underlying causes. These causes are quantified, in turn, by application of models and data on the respective unavailabilities due to hardware failure, common cause failure, human error, and test and maintenance unavailabilities. The frequencies of initiating events, the hardware failure rates of the components, and operator errors were obtained using either generic data or a combination of generic and plant-specific data.

Dependency matrices are developed from a detailed examination of the plant systems to account for important interdependencies and interactions that are highly plant specific. To facilitate a clear definition of plant conditions in the scenarios, separate stages of event trees are provided for the response of the support systems (e.g., electric power and cooling water), the frontline systems [e.g., high pressure coolant injection (HPCI) and residual heat removal (RHR)], and the containment phenomena; e.g., containment overpressurization failure. The containment event tree is used in the Level 2 PSA. A detailed definition of plant damage states provides the interface between the Level 1 and Level 2 event trees.

The systematic, structured approach that is followed in constructing the accident scenario model provides assurance that plant-specific features are identified. It also provides insights into the key risk controlling factors.