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TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

PROBABILISTIC SAFETY ASSESSMENT

UNIT 2 SUMMARY REPORT

Revision 1

ERIN®

ABS Consulting

RISK CONSULTING DIVISION

TVA

TENNESSEE VALLEY AUTHORITY
SYSTEMS AND ANALYSIS
BROWNS FERRY NUCLEAR PLANT

PROBABILISTIC SAFETY ASSESSMENT

UNIT 2 SUMMARY REPORT
Revision 1

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Browns Ferry Nuclear Plant
Probabilistic Safety Assessment
REVISION LOG

Unit 2 Summary Report

Revision No.	Description of Revision	Prepared By / Date	Checked By / Date	Approved By / Date
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TABLE OF CONTENTS

Section | Page

1.0 EXECUTIVE SUMMARY 1-1
- 1.1 BACKGROUND AND OBJECTIVES 1-1
- 1.2 PLANT FAMILIARIZATION 1-1
- 1.3 OVERALL METHODOLOGY 1-2
- 1.4 SUMMARY OF MAJOR FINDINGS 1-5
  - 1.4.1 Total Core Damage and Large Early Release Frequency 1-5
  - 1.4.2 Contributors to Total Core Damage Frequency 1-9
    - 1.4.2.1 Important Core Damage Sequence Groups 1-9
    - 1.4.2.2 Analysis of Individual Sequences 1-17
    - 1.4.2.3 Important Operator Actions 1-21
    - 1.4.2.4 Important Plant Hardware Characteristics 1-22
  - 1.4.3 Results for Large Early Release Frequency 1-23
    - 1.4.3.1 Important Plant Hardware Characteristics for Containment Performance 1-24
  - 1.4.4 COMPARISON WITH THE 2002 BROWNS FERRY UNIT 2 PRA, REVISION 0 1-24
- 1.5 INSIGHTS 1-25

2.0 REFERENCES 2-1

APPENDIX A Unit 2 Top Ranking Sequences Contributing to CDF

LIST OF TABLES

Table 1-1 Comparison with Other PRAs....1-8
Table 1-2 Unit 2 Initiating Event Group Contributions to Core Damage Frequency...1-11
Table 1-3 Summary of the Core Damage Accident Sequence Subclasses....1-15
Table 1-4 Breakdown of Core Damage Sequences in Each Frequency Range....1-17
Table 1-5 Browns Ferry Unit 2 Important Operator Actions....1-22
Table 1-6 Browns Ferry Unit 2 Important Systems....1-22
Table 1-7 Summary of Revised Human Error Probabilities....1-25

LIST OF FIGURES

Figure 1–1 Uncertainty Curve for Browns Ferry Unit 2 Core Damage Frequency....1-6
Figure 1–2 Uncertainty Curve for Browns Ferry Unit 2 Large Early Release Frequency....1-7
Figure 1-3 Browns Ferry Unit 2 Core Damage Frequency by Initiating Event Category....1-10

1.

SECTION 1
EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This documents the performance by the Tennessee Valley Authority (TVA) in revising the Unit 2 PSA. An integrated team of engineers and specialists from TVA and ABS Consulting performed this revision

TVAs overall objectives for this revision were to incorporate the Extended Power Uprate into the PSA.

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); data uncertainties in the estimated CDF; an estimate of the large early release frequency (LERF); and data uncertainties in the estimated LERF. This summary also provides the sequences, systems, and sources of uncertainty that are the significant contributors to the results.

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, with Unit 1 rated power level of 3,293 MWt and Unit 2 and 3 rated at 3,952 MWt. Unit 2 and Unit 3 are the only units 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 a Mark I drywell containment, 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 1-2).

1.3 OVERALL METHODOLOGY

The Browns Ferry Unit 2 PSA is founded on a scenario-based definition of risk (Reference 1-3). 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 that conveys the state of knowledge, or confidence, about that frequency.

Question 3 is answered in two ways. One measure is the core damage frequency. The loss of adequate core cooling is defined as the rapid increase in fuel clad temperature due to heating and Zircaloy-water reactions that lead to sudden deterioration of fuel clad integrity. For the purposes of the Level 1 PSA a surrogate has been developed that can be used as a first approximation to define the onset of core damage. The onset of core damage is defined as the time at which more than two-thirds of the active fuel becomes uncovered, without sufficient injection available to recover the core quickly, i.e., water level below one-third core height and falling. The other measure is the large, early release frequency. The original IPE answered question 3 in a Level 2 PSA, in terms of the key characteristics of radioactive material release that could result from the sequences identified. Consistent with recent PSA practice, BFN does not track the entire spectrum of releases. Instead, it tracks the frequency of large, early releases. A large early release is defined as the rapid, unscrubbed release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions. 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 1-4).

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 1-5) was used to construct effectively a single, large tree for Level 1 and LERF. The sequences analysis start with an initiating event and terminate in end states of LERF or no LERF. The sum of these two end states is the CDF.

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 have previously been 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. A separate tree is used to determine core damage and develop plant damage classes. This tree 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 presentation describes both the core damage and large early release results.

1.4.1 Total Core Damage and Large Early Release Frequency

The total CDF for Browns Ferry Unit 2 was found to be 2.7×10^{-6} per reactor-year. The results for CDF were developed in terms of a mean point estimate. The CDF data uncertainty curve is shown in Figure 1-1.

The total Large Early Release Frequency (LERF) for Browns Ferry Unit 2 was found to be 4.0×10^{-7} per reactor year. The results for LERF were developed in terms of a mean point estimate. The LERF data uncertainty curve is shown in Figure 1-2.

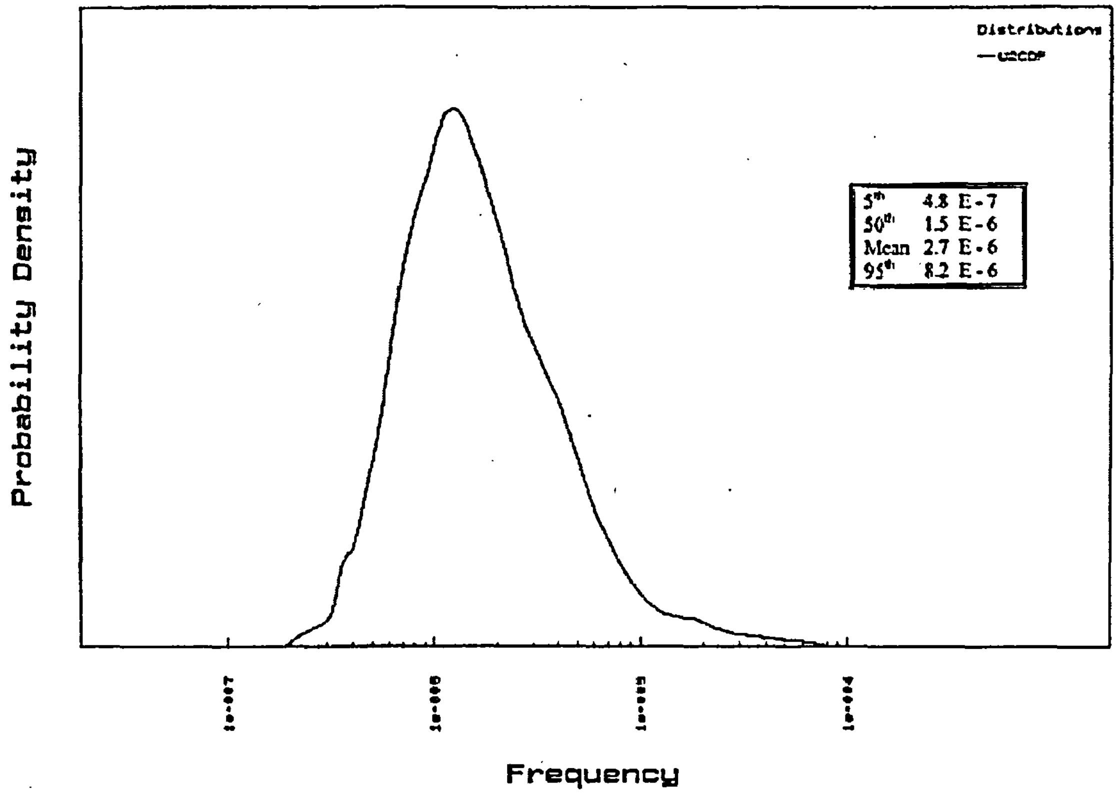

Figure 1-1
Uncertainty Curve for Browns Ferry Unit 2 Core Damage Frequency

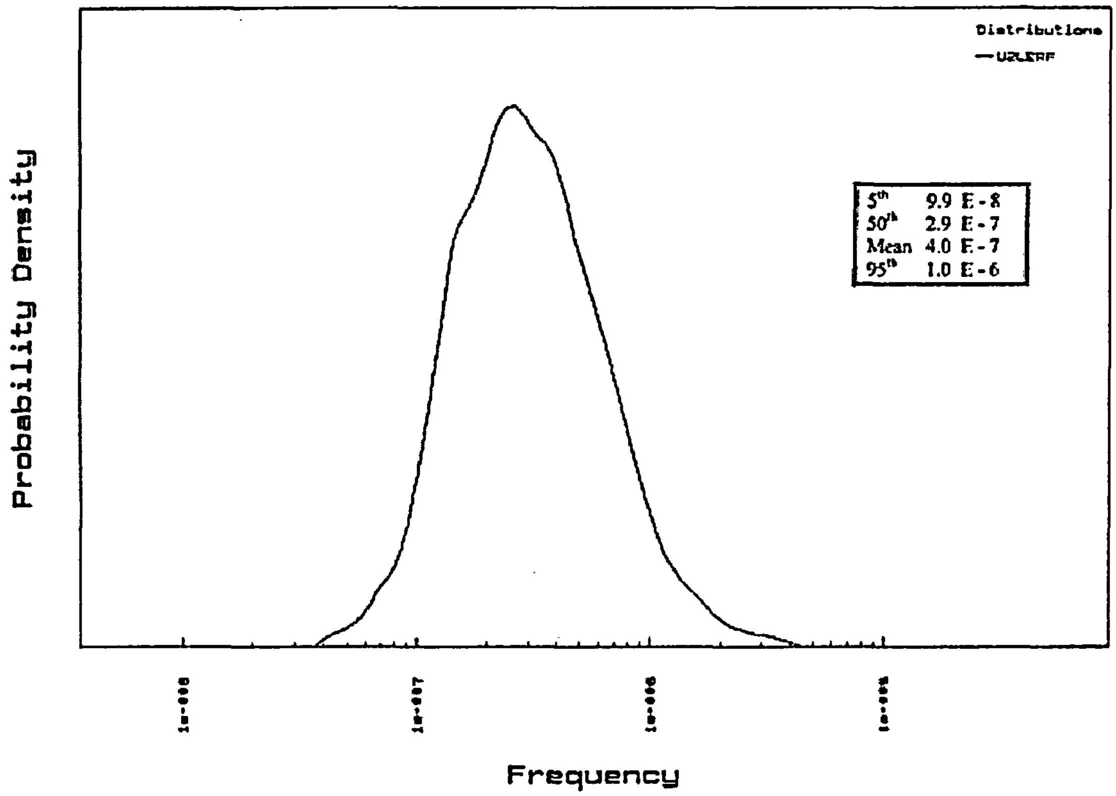

Figure1-2
Uncertainty Curve for Browns Ferry Unit 2 Large Early Release Frequency

A comparison of this study with other PSAs on other plants that used similar methods, databases, and work scopes is given in Table 1-1. The calculated mean CDF for Browns Ferry Unit 2 is of the same order of magnitude as Quad Cities, Peach Bottom Unit 2 and Grand Gulf Unit 1, and an order of magnitude lower than that reported for Nine Mile Point Unit 2 (which includes external events).

Table 1-1

Comparison with Other PRAs

Plant	Flood Included	Mean CDF (per reactor-year)	Reference	Mean LERF (per year)
Quad Cities	Yes	4.6E-6	1-7	3.3E-6
Nine Mile Point Unit 2*	Yes	5.7E-5	1-8	1.6E-6
Browns Ferry Unit 2	Yes	2.7E-6	This Study	4.0E-7
Peach Bottom Unit 2	No	4.5E-6	1-9	Not Updated
Grand Gulf Unit 1	No	5.5E-6	1-10	Not Updated

*Includes external events.

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

- The increase in core thermal power resulting from the EPU eliminated the use of CRD as an alternative injection source if the vessel remains at high pressure and other injection sources fail. The increase in the CDF estimate from Revision 0 is largely due to the elimination of this success path.

- The accident sequences that were analyzed are those initiated by internal events and internal floods. Sequences initiated by internal fires, seismic events, and other external events have not been modeled in this internal events model.

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

- This study used plant specific data to update failure rates for selected components and initiating events frequencies. The common cause parameters of the multiple Greek model used in this study were estimated with the benefit of a plant-specific screening of industry common cause event data in accordance with NUREG/CR-4780 (Reference 1-11). The common cause event data was taken from the NRC database (Reference 1-14). Common cause estimates not screened were taken from NUREG/CR-5497.

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. 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 1-3 and Table 1-2 for major initiating event categories.

Transients with the Power Conversion System (PCS) unavailable as a result of the initiator account for 32.9% of the CDF. Loss of condenser heat sink, which includes closure of the main steam isolation valves and turbine trip without bypass, are specific examples of initiator in this group.

Transients with the PCS not disabled as a result of the initiator contribute 29.5% to the core damage frequency. The turbine trip, in which the main steam isolation valves and turbine bypass are available, is a specific example of an initiator in this group.

The Loss of Offsite Power (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 pressure injection) and decay heat removal failures. Overall, the LOSP initiated sequences account for 18% of CDF.

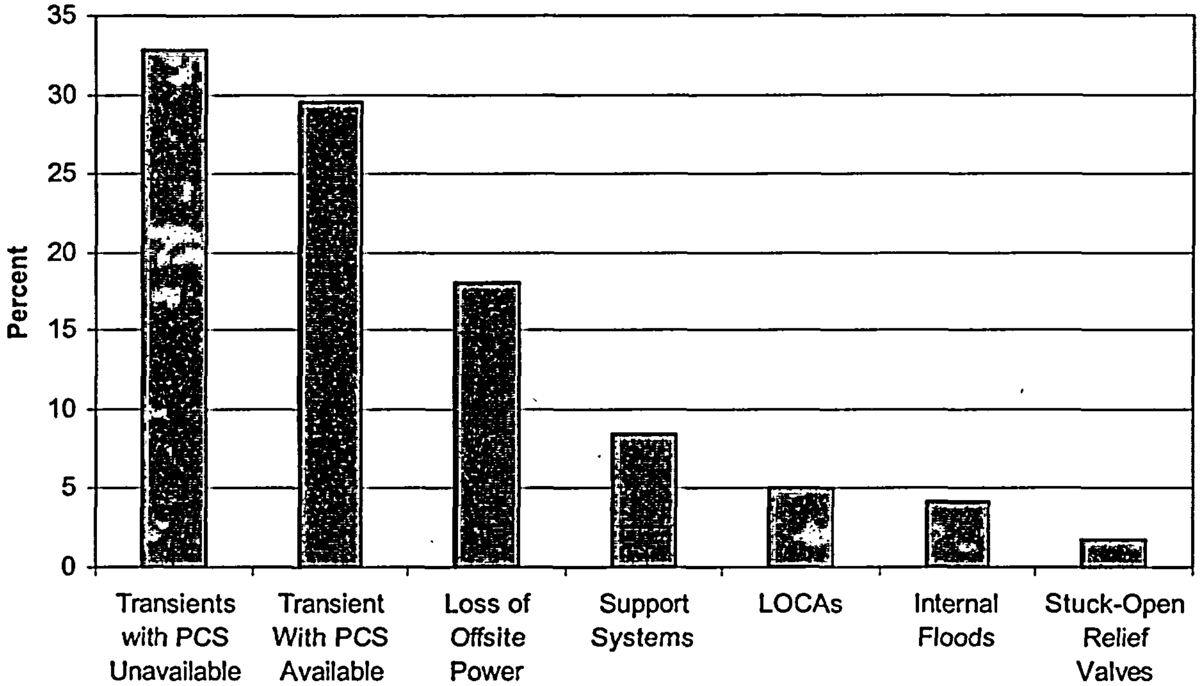

Figure 1-3
Browns Ferry Unit 2 Core Damage Frequency by Initiating Event Category

Table1-2

Unit 2 Initiating Event Group Contributions to Core Damage Frequency

Initiating Event Category	Mean CDF (per reactor-year)	Percentage of Total
Transients with PCS Unavailable	8.79E-07	32.9
Transients with PCS Available	7.90E-07	29.5
Loss of Offsite Power	4.83E-07	18.0
Support System Failures	2.26E-07	8.5
LOCAs	1.37E-07	5.1
Internal Floods	1.12E-07	4.2
Stuck-Open Relief valves	4.83E-08	1.8
Total	**2.7E-06**	100

Support system failure initiators (specifically, loss of plant air, loss of raw cooling water, or loss of either I&C bus 2A or 2B failures) contribute 8.5% to the total CDF.

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 5.1% of the total CDF.

Scenarios initiated by internal floods contribute 4.2% 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, and in the turbine building (two sizes).

Scenarios initiated by the inadvertent opening of one or more safety relief valves (SRVs) contribute 1.8% the core damage frequency. Two distinct initiators are considered: opening of one SRV, and opening of two or more SRVs.

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.

An event sequence classification into five accident sequence functional classes can be performed using the functional events as a basis for selection of end states. The description of functional classes is presented here to introduce the terminology to be used in characterizing the basic types of challenges to containment. The reactor pressure vessel condition and containment condition for each of these classes at the time of initial core damage is noted below:

Core Damage Functional Class	RPV Condition	Containment Condition
I	Loss of effective coolant inventory (includes high and low pressure inventory losses)	Intact
II	Loss of effective containment pressure control, e.g., heat removal	Breached or Intact
III	LOCA with loss of effective coolant inventory makeup	Intact
IV	Failure of effective reactivity control	Breached or Intact
V	LOCA outside containment	Breached (bypassed)

In assessing the ability of the containment and other plant systems to prevent or mitigate radionuclide release, it is desirable to further subdivide these general functional categories. In the second level binning process, the similar accident sequences grouped within each accident functional class are further discriminated into subclasses such that the potential for system recovery can be modeled. These subclasses define a set of functional characteristics for system operation which are important to accident progression, containment failure, and source term definition. Each subclass contains front end sequences with sufficient similarity of system functional characteristics that the containment accident progression for all sequences in the group can be considered to behave similarly in the period after core damage has begun. Each subclass defines a unique set of conditions regarding the state of the plant and containment systems, the physical state of the core, the primary coolant systems, and the containment boundary at the time of core damage, as well as vessel failure.

The important functional characteristics for each subclass are determined by defining the critical parameters or system functions that impact key results. The sequence characteristics that are important are defined by the requirements of the containment accident progression analysis. These include the type of accident initiator, the operability of important systems, and the value of important state variables (e.g., reactor pressure) that are defined by system operation. The interdependencies that exist between plant system operation and the core melt and radionuclide release phenomena are represented in the release frequencies through the binning process involving these subclasses, as shown in past PRAs and PRA reviews. The binning process, which consolidates information from the systems' evaluation of accident sequences leading to core damage in preparation for transfer to the containment-source term evaluation, involves the identification of 13 classes and subclasses of accident sequence types. Table 1-3 provides a description of these subclasses that are used to summarize the Level 1 PRA results.

Published BWR PRAs have identified that there may be a spectrum of potential contributors to core melt or containment challenge that can arise for a variety of reasons. In addition, sufficient analysis has been done to indicate that the frequencies of these sequences are highly uncertain; and therefore, the degree of importance on an absolute scale and relative to each other, depends upon the plant specific features, assumptions, training, equipment response, and other items that have limited modeling sophistication.

This uncertainty means that the analyst can neither dismiss portions of the spectrum from consideration nor emphasize a portion of the spectrum to the exclusion of other sequence types. This is particularly true when trying to assess the benefits and competing risks associated with a modification of a plant feature.

This end state characterization of the Level 1 PRA in terms of accident subclasses is usually sufficient to characterize the CET entry states for most purposes. However, when additional refinement is required in the CET quantification, it may be useful to further discriminate among the contributors to the core damage accident classes. This discrimination can be performed through the use of the individual accident sequence characteristics.

Table1-3
Summary of the Core Damage Accident Sequence Subclasses

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	TQUX
	B	Accident sequences involving a station blackout and loss of coolant Inventory makeup.	T_EQUV
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	T_TC_MQU
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi; i.e., accident sequences initiated by common mode failures disabling multiple systems (ECCS) leading to loss of coolant inventory makeup.	TQUV
	E	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high and DC power is unavailable.	---
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure	TW
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage after containment failure.	AW
	T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post high containment pressure	N/A
	V	Class IIA or IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	TW

Table1-3
Summary of the Core Damage Accident Sequence Subclasses

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class III LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	R
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	S,QUX
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	AV
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	AD
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	$T_TC_MC_2$
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g., LOCA or SORV); core damage induced post containment failure.	N/A
	T	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post high containment pressure.	N/A
	V	Class IV A or L except that the vent operates as designed, loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	N/A
Class V		Unisolated LOCA outside containment	N/A

For BFN, functional based plant damage states are used to summarize Level 1 results and to ensure that the Level 2 CETs are sufficient to allow each functional sequence to be addressed.

1.4.2.2 Analysis of Individual Sequences

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
>1E-07	1	6
>1E-08	29	30
>1E-09	298	57
>1E-10	2545	78
>1E-11	18,089	96
>1E-12	18,089 + Not saved	100

The following presents a brief description of the 20 highest-ranking sequences to the CDF.

A loss of condenser heat sink initiates the first sequence. The initiator directly causes a loss of reactor feedwater, degrading high pressure injection capabilities. Subsequent failures of HPCI and RCIC eliminate all of high pressure injection. The remaining success path of low pressure injection is not viable because of a failure to depressurize. A lack of inventory causes core damage.

The second sequence is similar to the first. It differs in that the end state of the second sequence is a large early release (LERF).

A general transient initiates the third sequence. A subsequent loss of the main condenser results in a situation identical to the first sequence initiator, a loss of the condenser heat sink. The remainder of sequence three is identical to that of sequence one.

The fourth sequence is that of an interfacing system LOCA that results in core damage. This sequence represents the total contribution from a variety of interfacing system LOCAs. An interfacing system LOCA is initiated by leakage of reactor coolant through valves that separate the nuclear boiler from the RHR or core spray systems.

Sequence five is the classic SBO following a total LOSP. The unit 1/2 diesel generators fail and the Unit 3 diesel generators fail by common cause. Offsite power is not recovered before core damage occurs.

Sequence six is initiated by a loss of offsite power to Unit 2. Although the diesel generators are successful, this sequence progresses to core damage as HPCI and RCIC fail, followed by a failure to depressurize.

A general transient initiates sequence seven. It is similar to sequence three but the failure of Reactor feedwater is caused by the failure of the turbine bypass valves.

Sequence eight is a non-minimal version of sequence one. The additional failure is Unit 3 at power.

Sequence nine is similar to sequence two but represents a different path in the LERF event tree.

A general transient initiates sequence ten. It is similar to sequence three but the failure of reactor feedwater is caused by the failure of the turbine bypass valves.

Sequence eleven is similar to sequence five. There is a logic error that fractures sequence five. The sum of sequence five and sequence eleven is correct.

Sequence twelve is also initiated by a loss of offsite power to both units. The combination of failures of emergency diesel generators results in failure of the EECW, which causes core damage due to the failure of components requiring EECW. Note that there are four trains of RHRSW/EECW with the number 3 pump in each train dedicated to EECW. The number 2 pump in each train is dedicated to RHR heat exchangers, and the number 1 pump in each train aligned as a back up the number 2 pump but capable of being aligned to backup the number 1 pump. Furthermore, trains C and D can be realigned via MOVs but trains A and B use manual values that require local operation. In this scenario DGs B, C, 3EA, and 3EB also fail. The failure of DG B and DG 3EB fail the C train. The failure of DG 3A fails the A3 EECW pump. The failure of DG C fails the B3 EECW pump. Three EECW pumps have failed and the success criteria for EECW are that two pumps are required. Since the A and C realignment is local, the EECW system cannot be restored before the DGs fail on loss of cooling. Equipment fails due to lack of EECW and offsite power is not recovered in time.

Sequence thirteen is similar to sequence twelve except that different combinations of diesel generators fail such that the EECW success criterion is not met. In this case, the C, D, 3EA, and 3ED diesel generators fail.

Sequence fourteen is a non-minimal version of sequence three.

Sequence fifteen is a non-minimal version of sequence two.

A flood in the turbine building initiates sequence sixteen. The flood disables feedwater. The remainder of the sequence is the familiar failure of high pressure injection with a failure to depressurize.

Sequence seventeen is a non-minimal version of sequence six.

Sequence eighteen is initiated by a loss of all condensate. This causes a loss of feedwater. HPCI and RCIC fail followed by a failure to depressurize.

Sequence nineteen is initiated by loss of reactor building closed cooling water and is then similar to sequence eighteen.

Sequence twenty is an LOSP initiator followed by a failure of the Unit 2 diesel generators and diesel generator 3ED. This combination results in a loss of cooling to the RHR heat exchangers and fails the cross-ties to Units 1 and 3. Offsite power is not recovered within 6 hours.

Section Appendix A contains a listing of the top 50 sequences.

The table below shows the frequency, percentage of total, and the cumulative percentage of total for the sequences discussed above.

Sequence	Frequency	% CDF	Cumulative
1	1.58E-07	5.84E-02	5.84E-02
2	5.84E-08	2.16E-02	8.00E-02
3	5.76E-08	2.13E-02	1.01E-01
4	4.63E-08	1.72E-02	1.18E-01
5	4.56E-08	1.69E-02	1.35E-01
6	4.38E-08	1.62E-02	1.52E-01
7	2.96E-08	1.10E-02	1.63E-01
8	2.77E-08	1.03E-02	1.73E-01
9	2.44E-08	9.02E-03	1.82E-01
10	2.42E-08	8.96E-03	1.91E-01
11	2.39E-08	8.86E-03	2.00E-01
12	2.29E-08	8.49E-03	2.08E-01
13	2.29E-08	8.49E-03	2.17E-01

14	2.13E-08	7.90E-03	2.25E-01
15	1.75E-08	6.49E-03	2.31E-01
16	1.68E-08	6.21E-03	2.37E-01
17	1.62E-08	6.01E-03	2.43E-01
18	1.60E-08	5.94E-03	2.49E-01
19	1.43E-08	5.29E-03	2.55E-01
20	1.41E-08	5.23E-03	2.60E-01

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.

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. No other HEPs are shown because of a dramatic fall off in importance.

Table1-5
Browns Ferry Unit 2 Important Operator Actions

Operator Action	Top Event/Split Fraction	Importance
Depressurize to Allow Low Pressure Injection	ORVD2	55.4
Open the Hardened Wetwell Vent	OLP4	12.6
Align Alternate Injection to Reactor Vessel via the Unit 3 to Unit 2 RHR Crosstie*	U32A	7.4
Operator Aligns Suppression Pool Cooling	OSP	5.2
*The importance of the split fraction U32 was weighted by the relative contribution of the human action contained in the system analysis		

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
HPCI	63
RCIC	57
Feedwater/Condensate	27
Diesel Generators	17
Main Steam	11
RPS	9
RHR	4
Control Rod Drive	2
RHRSW to RHR Loop II	1
RHRSW to RHR Loop I	1
Standby Liquid Control	1
Core Spray	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.

1.4.3 Results for Large Early Release Frequency

This section summarizes the limited results for the Level 2 analysis, which estimates the large containment failure and subsequent early release of radionuclides known as LERF. In contrast to the IPE submittal, this update concerned itself with two metrics, core damage frequency and large early release frequency. This section presents the LERF results and contributors.

The results indicate that about 15% of the core damage scenarios result in LERF. Typically, LERF as a percentage of CDF for BWRs ranges from 10% to almost 50%. These are generally highly dependent on the level 1 results. BFN Unit 2 falls in the mid-range for BWRs.

This release category represents the most severe source term scenario. Here the containment failures are dominated by drywell shell failures (due to thermal interactions with the molten core debris on the drywell floor). The important post-core damage contributors are drywell shell failures, in-vessel recovery, and the effectiveness of the reactor building in scrubbing the release. With respect to pre-core damage top events, the failure of the RPS system dominates.

1.4.3.1 Important Plant Hardware Characteristics for Containment Performance

As discussed in the previous Section 1.4.3.1, the dominant contributor to the most significant release category group (large, early containment failure and large bypasses) is drywell shell thermal attack from corium on the drywell floor. This is representative for most Mark I containments. The likelihood of drywell shell thermal attack failure is significantly reduced if the drywell floor is flooded with water prior to vessel breach. Drywell spray represents an important hardware component since it is the primary means of flooding the drywell.

1.4.4 COMPARISON WITH THE 2002 BROWNS FERRY UNIT 2 PRA, REVISION 0

TVA has previously performed an individual plant examination in accordance with the U.S. Nuclear Regulation Commission (NRC) Generic Letter No. 88-20 (Reference 1-1). The IPE was revised on several occasions. PSA Revision 0 marked the change from IPE to an application and risk informed approach. This Revision 1 reflects plant operations with the extended power uprate. The increase in thermal power eliminated the use of the CRD system as an effective injection source when the vessel remains at high pressure and the other high pressure injection sources have failed. The increase in thermal power also required revisions to some human actions due to the change in sequence timing. See Table 1-7.

Table 1-7
Summary of Revised Human Error Probabilities

Operator Action	CPPU HEP	Current HEP	Description
HOAD1	4.89E-03	3.45E-03	Inhibit ADS During ATWS with Unisolated Vessel
HOAD2	9.52E-03	4.64E-03	Inhibit ADS During ATWS with Isolated Vessel
HOAL2	1.29E-01	3.91E-02	Lower and Control Vessel Level
HOSL1	1.61E-02	6.71E-03	Initiate SLCS Given ATWS with Unisolated RPV.
HOSL2	7.71E-02	3.50E-02	Initiate SLCS, Given an ATWS with RPV Isolated

1.5 INSIGHTS

The power increase eliminated the use of CRD as a viable high pressure injection if the vessel remains at high pressure. The increase in CDF given EPU as compared to the current model is almost entirely due to this elimination. The high pressure injection systems and the operator action to depressurize are much more important given EPU.

It is noted that LOSP initiated sequences are of higher frequency for Unit 3 than for Unit 2. This is due to the different board layouts and resulting dependencies between the units. On Unit 3, the failure of 3 DGs (and associated boards) will fail all the RHR pumps. Failure of DGs 3EA, 3EB, and 3EC fail the motive power for RHR pumps A, B, and C, and fails 480V shutdown board 3B. 480V shutdown board 3B supplies room cooling to the Unit 3 B and D RHR pumps. HPCI fails long term because of the failure of DG 3EA, which maintains the charger for long-term 250V DC power. This is the trigger for the higher frequency LOSP sequences on Unit 3. Core damage results with a failure of RCIC.

The fact that RCIC long-term operation requires both DGs A and B aggravates the situation.

In contrast, no combination of 3 Unit 2 DG failures will guarantee the failure of Unit 2 RHR. Further, the Unit 2 RCIC does not depend on the Unit 3 boards.

SECTION 2

REFERENCES

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1-3. Kaplan, S., and Garrick, B. J., "On the Quantitative Definition of Risk," *Risk Analysis,* Vol. 1, pp. 11-37, March 1981.

1-4. American Nuclear Society and Institute of Electrical and Electronics Engineers, "PRA Procedures Guide; A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," sponsored by U.S. Nuclear Regulatory Commission and Electric Power Research Institute, NUREG/CR-2300, April 1983.

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1-6. Deleted

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1-10. Conversation between Shawn S. Rodgers, ERIN Engineering and Research, Inc. and Gary Smith, Entergy, May 17, 2000.

1-11. Mosleh, A., et al., "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," Pickard, Lowe and Garrick, Inc., prepared for U.S. Nuclear Regulatory Commission and Electric Power Research Institute, NUREG/CR-4780, EPRI NP-5613, PLG-0547, Vols. 1-2, January 1988.

1-12. Deleted.

1-13. Deleted.

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APPENDIX A

UNIT 2 TOP RANKING SEQUENCES CONTRIBUTING TO CDF

1	LOCHS	207	1.5761E-007	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FWNLERF AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF
2	LOCHS	211	5.8376E-008	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FW NLERF AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF*FC2*RBEF
3	TRAN	543	5.7617E-008	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FW NLERF AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF
4	ISLOCA	1	4.6342E-008	
5	LOSP	3237	4.5649E-008	OG5F*OG16F*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*G NLERF A1*GD2*GB4*GC4*EPR303*DGC1*AAF*RQF*REF*RMF*ABF*RSF*RH F*UB42CF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RIF*RJF*RNF* DLF*DOF*UB43AF*UB43BF*GEF*A3EAF*RXF*ROF*DNF*GGF*A3ECF *GFF*A3EBF*RYF*RPF*DWF*RCWF*EAF*EBF*EC F*EDF*RBCF*SW2AF*SW1AF*SW2BF*SW1BF*SW2CF*SW1CF*SW2DF* PCAF*DCAF*CADF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWA F*HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*R480F*RPAF*RPCF*U1F*R PBF*RPDF*U3F*OSPF*LPCF*OAIF*NCDF*RHSWF*NOCDF
6	L500U2	77	4.3814E-008	OG5F*SDRECF*OXF*DWF*MCDF*RVC0*FWHF*RCI1*HPI4*OBDF*ORV NLERF D2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*EL F*WWBF*WWF
7	TRAN	915	2.9570E-008	SDRECF*OXF*DWF*TB1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWA NLERF F*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF *WWF
8	LOCHS	1213	2.7727E-008	SDRECF*OXF*DWF*U3AP1*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*OR NLERF VD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*E LF*WWBF*WWF
9	LOCHS	209	2.4361E-008	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FW NLERF AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF*RBI2
10	TRAN	209	2.4201E-008	SDRECF*OXF*DWF*BVR1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FW NLERF AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF
11	LOSP	3313	2.3927E-008	OG5F*OG16F*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*G NLERF A1*GD2*GB4*GC4*EPR303*DGC1*AAF*RQF*REF*RMF*ABF*RSF*RH F*UB42CF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RIF*RJF*RNF* DLF*DOF*UB43AF*UB43BF*GEF*A3EAF*RXF*ROF*DNF*GGF*A3ECF *GFF*A3EBF*RYF*RPF*GH7*A3EDF*SDRECF*DW F*RCWF*EAF*EBF*ECF*EDF*RBCF*SW2AF*SW1AF*SW2BF*SW1BF*S W2CF*SW1CF*SW2DF*SW1DF*PCAF*DCAF*CADF*OEEF*IVOF*RVC0* CDF*EPR63*RCLF*HPLF*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*ORP F*R480F*RPAF*RPCF*U1F*RPBF*RPDF*U3F*OSPF*LPCF*OAIF*NC DF*RHSWF*NOCDF
12	LOSP	1116	2.2935E-008	OG5F*OG16F*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*G NLERF B1*GC2*EPR303*ABF*RSF*RHF*UB42CF*DKF*ACF*RRF*RFF*UB43 AF*UB43BF*GE1*A3EAF*RXF*ROF*DNF*GF2*A3EBF*SDRECF*DWF* RCWF*EAF*EBF*ECF*RBCF*SW2BF*SW2CF*SW1CF*PCAF*DCAF*OEE F*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*H RLF*SUFWF*HSF*CDAF*CRDF*ORPF*RPAF*RPCF*U1F*RPBF*RPDF* U3F*OSPF*LPCF*OAIF*NCDF*RHSWF*NOCDF
13	LOSP	1556	2.2927E-008	OG5F*OG16F*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*G NLERF D1*GC2*EPR303*UB42CF*ACF*RRF*RFF*ADF*RTF*RIF*RJF*RNF* DLF*DOF*UB43AF*UB43BF*GE1*A3EAF*RXF*ROF*GH2*A3EDF*SDR ECF*DWF*RCWF*EAF*EBF*EDF*RBCF*SW2BF*SW2DF*SW1DF*PCAF* DCAF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPL F*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*RPAF*RPCF*U1F*RP BF*RPDF*U3F*OSPF*LPCF*OAIF*NCDF*RHSWF*NOCDF
14	TRAN	547	2.1340E-008	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FW NLERF AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB

				F*WWF*FC2*RBEF	
15	LOCHS	208	1.7513E-008	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FW AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF*RBE4	NLERF
16	FLTB2	90	1.6760E-008	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*ORVD2*FWAF*HRL F*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
17	L500U2	81	1.6227E-008	OG5F*SDRECF*OXF*DWF*MCDF*RVC0*FWHF*RCI1*HPI4*OBDF*ORV D2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*EL F*WWBF*WWF*FC2*RBEF	NLERF
18	LOAC	50	1.6038E-008	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*ORVD2*FWAF*HRL F*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
19	LRBCCW	75	1.4274E-008	SDRECF*OXF*DWF*RBCF*DCAF*IVOF*RVC0*FWHF*RCI1*HPI4*OIV F*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLER FF*ELF*WWBF*WWF	NLERF
20	LOSP	3045	1.4118E-008	OG5F*OG16F*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*G A1*GD2*GB4*GC4*EPR303*AAF*RQF*REF*RMF*ABF*RSF*RHF*UB4 2CF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RIF*RJF*RNF*DLF*D OF*UB43AF*UB43BF*GH1*A3EDF*DWF*RCWF*EBF*RBCF*SW2AF*SW 1AF*SW2BF*SW2CF*SW1CF*SW2DF*SW1DF*PCAF *DCAF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF*SUFWF*H SF*CDAF*CRDF*R480F*RPAF*RPCF*U1F*RPBF*RPDF*U3F*OSPF*L PCF*OAIF*NCDF*RHSWF*NOCDF	NLERF
21	LOPA	20	1.3638E-008	SDRECF*OXF*DWF*PCAF*DCAF*IVOF*RVC0*FWHF*RCI1*HPI4*OIV F*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*LCF*NCDF*NOCDF* NLERFF*ELF*WWBF*WWF	NLERF
22	TRAN	1668	1.2162E-008	SDRECF*OXF*DWF*RXS1*OSL1*NAF*FWAF*HRLF*HR6F*SUFWF*CDA F*NCDF*NOCDF*NLERFF*CILF*IVR10*TR6*FCF*DWIF*RBE7	LERF
23	TRAN	1677	1.2162E-008	SDRECF*OXF*DWF*RXS1*OSL1*NAF*FWAF*HRLF*HR6F*SUFWF*CDA F*NCDF*NOCDF*NLERFF*CILF*WW1*IVR10*TR6*FCF*DWIF*RBE8	LERF
24	L500PA	27	1.1938E-008	OG5F*SDRECF*OXF*DWF*MCDF*RVC0*FWHF*RCI1*HPI4*OBDF*ORV D2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*EL F*WWBF*WWF	NLERF
25	BOC	16	1.1808E-008	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCIF*HPI4*OIVF*ORVD2*FW AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF	NLERF
26	TRAN	919	1.0952E-008	SDRECF*OXF*DWF*TB1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWA F*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF *WWF*FC2*RBEF	NLERF
27	ISCRAM	174	1.0355E-008	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FW AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF	NLERF
28	LOCHS	1217	1.0269E-008	SDRECF*OXF*DWF*U3AP1*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*OR VD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*E LF*WWBF*WWF*FC2*RBEF	NLERF
29	TRAN	2795	1.0136E-008	SDRECF*OXF*DWF*U3AP1*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*OR VD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*E LF*WWBF*WWF	NLERF
30	LRCW	183	9.5325E-009	SDRECF*OXF*DWF*RCWF*MCDF*RVC0*CDF*RCI1*HPI4*ORVD2*FWA F*HRLF*HR6F*SUFWF*HSF*CDAF*CRDF*NCDF*NOCDF*NLERFF*ELF *WWBF*WWF	NLERF
31	LOCHS	231	9.5135E-009	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FW AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF*OP3*IVR1*RBE5	LERF
32	ELOCA	1	9.3900E-009	NCDF	LERF
33	LOCHS	212	9.0226E-009	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FW AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF*FC2*RBI2*RBEF	NLERF
34	TRAN	213	8.9634E-009	SDRECF*OXF*DWF*BVR1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FW AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF*FC2*RBEF	NLERF

35	LOCHS	241	8.9143E-009	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*RHSW1*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
36	TRAN	545	8.9053E-009	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF*RBI2	NLERF
37	TRAN	2187	8.8426E-009	SDRECF*OXF*DWF*DCA1*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
38	LOSP	3280	7.9802E-009	OG5F*OG16F*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GA1*GD2*GB4*GC4*EPR303*DGC1*AAF*RQF*REF*RMF*ABF*RSF*RHF*UB42CF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RIF*RJF*RNF*DLF*DOF*UB43AF*UB43BF*GEF*A3EAF*RXF*ROF*DNF*GGF*A3ECF*GFF*A3EBF*RYF*RPF*DWF*U3AP1*RCWF*EAF*EBF*ECF*EDF*RBCF*SW2AF*SW1AF*SW2BF*SW1BF*SW2CF*SW1CF*PCAF*DCAF*CADF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*R480F*RPAF*RPCF*U1F*RPBF*RPDF*U3F*OSPF*LPCF*OAIF*NCDF*RHSWF*NOCDF	NLERF
39	TRAN	383	7.8836E-009	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*OBCF*FWAF*HSF*HXA1*HXC2*U12*HXB5*HXD7*U32A*OSPF*OSDF*OLP2*NCDF*NOCDF	NLERF
40	LOFW	47	7.7907E-009	SDRECF*OXF*DWF*RVC0*FWHF*RCI1*HPI4*OBD1*ORVD2*FWAF*HRLF*HR6F*SUFWF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
41	L500U2	495	7.7076E-009	OG5F*SDRECF*OXF*DWF*U3AP1*MCDF*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
42	LOCHS	490	7.3326E-009	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*CRD11*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
43	TRAN	364	7.1699E-009	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*OBCF*FWAF*HSF*OSP1*SDC2*OLP2*NCDF*NOCDF	NLERF
44	TRAN	3416	6.9274E-009	DJ1*SDRECF*OXF*DWF*TBF*RVC0*FWHF*RCIF*HPI6*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*LCF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
45	LOSP	39	6.9202E-009	OG5F*OG16F*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*UB42CF*SDRECF*DWF*RCWF*IVOF*RVC0*CDF*RCI1*HPI4*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*CRDF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
46	L500U2	79	6.7719E-009	OG5F*SDRECF*OXF*DWF*MCDF*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF*RBI2	NLERF
47	TRAN	544	6.4019E-009	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF*RBE4	NLERF
48	FLTB2	94	6.2074E-009	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF*FC2*RBEF	NLERF
49	LOAC	54	5.9399E-009	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF*FC2*RBEF	NLERF
50	LOCHS	73	5.6349E-009	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*OHC1*L8H1*ORVD3*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF