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Bob,
Attached is the Unit 3 PSA Summary Report.

Gary M. Adkins, P.E.
BFN License Renewal Manager
Phone:(423)751-4363

<<Unit 3 PSA Summary Report Rev 1.pdf>>

CC: "Morrison, George M." <gmmorrison@tva.gov>

TENNESSEE VALLEY AUTHORITY
SYSTEMS AND ANALYSIS
BROWNS FERRY NUCLEAR PLANT

PROBABILISTIC SAFETY ASSESSMENT

UNIT 3 SUMMARY REPORT
Revision 1

January 2003

Browns Ferry Nuclear Plant
Probabilistic Safety Assessment
REVISION LOG

Unit 3 Summary Report

Revision No.	Description of Revision	Prepared By / Date	Checked By / Date	Approved By / Date
0	Initial Issue	D. Bidwell	S. Rodgers	D.McCamy/
1	Changes from EPU	S. Rodgers	D. Johnson	

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1.

SECTION 1
EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This documents the performance by the Tennessee Valley Authority (TVA) in revising the Unit 3 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 3. 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 3 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 3 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 3 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 3.

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. Core damage is the loss of adequate core cooling 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 3 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 3 was found to be 3.4×10^{-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 3 was found to be 4.5×10^{-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.

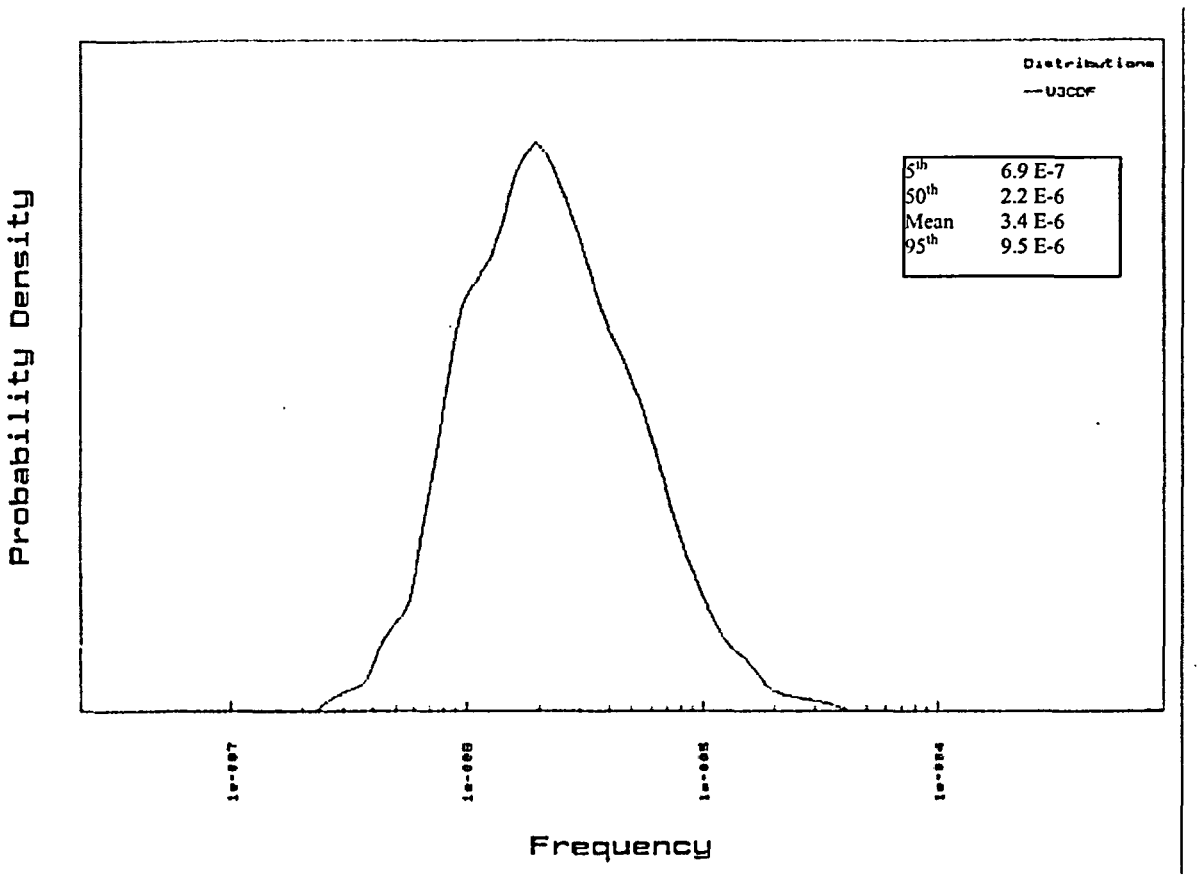

Figure 1-1

Uncertainty Curve for Browns Ferry Unit 3 Core Damage Frequency

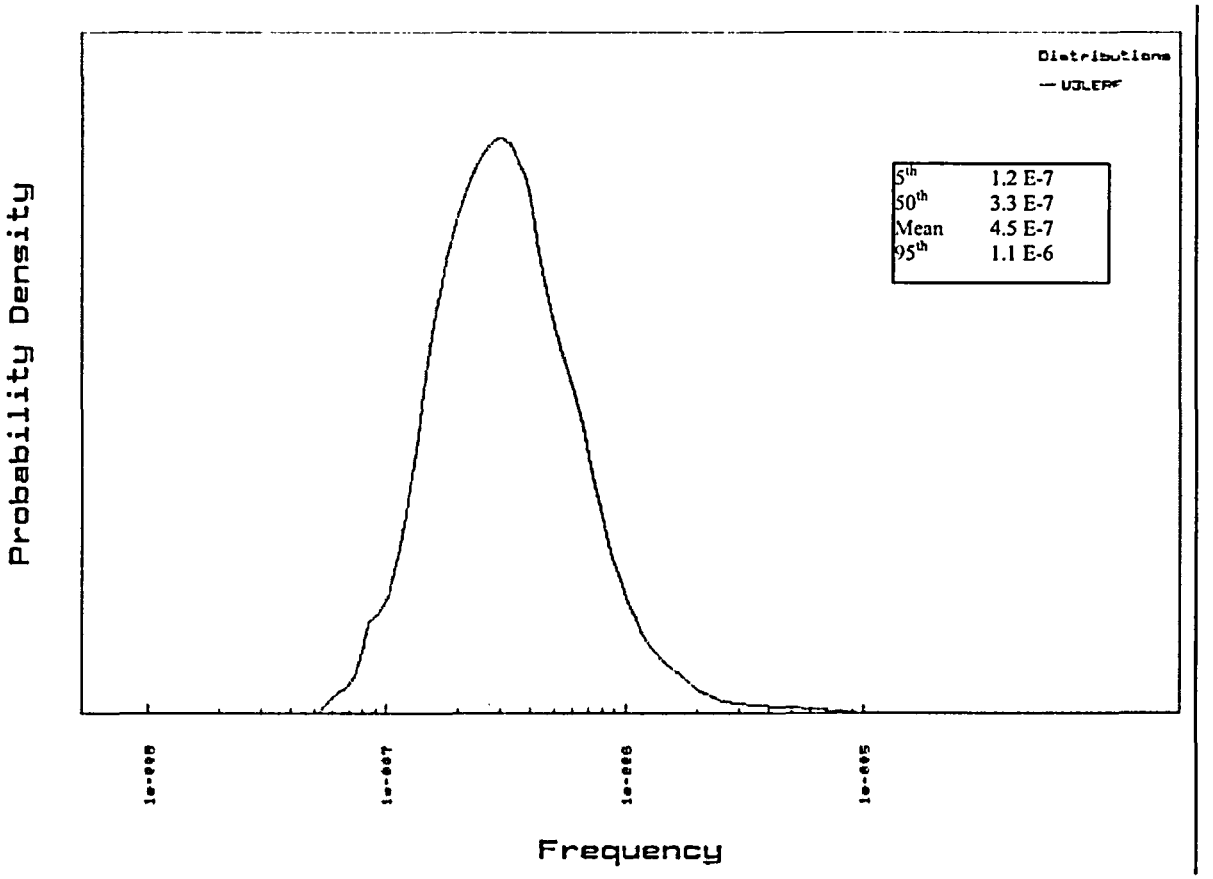

Figure 1-2

Uncertainty Curve for Browns Ferry Unit 3 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 3 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 3	Yes	3.4E-6	This Study	4.5E-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 3 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.
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 31.3% of CDF.

Transients with the Power Conversion System (PCS) unavailable as a result of the initiator account for 28.3% 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 23.8% 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.

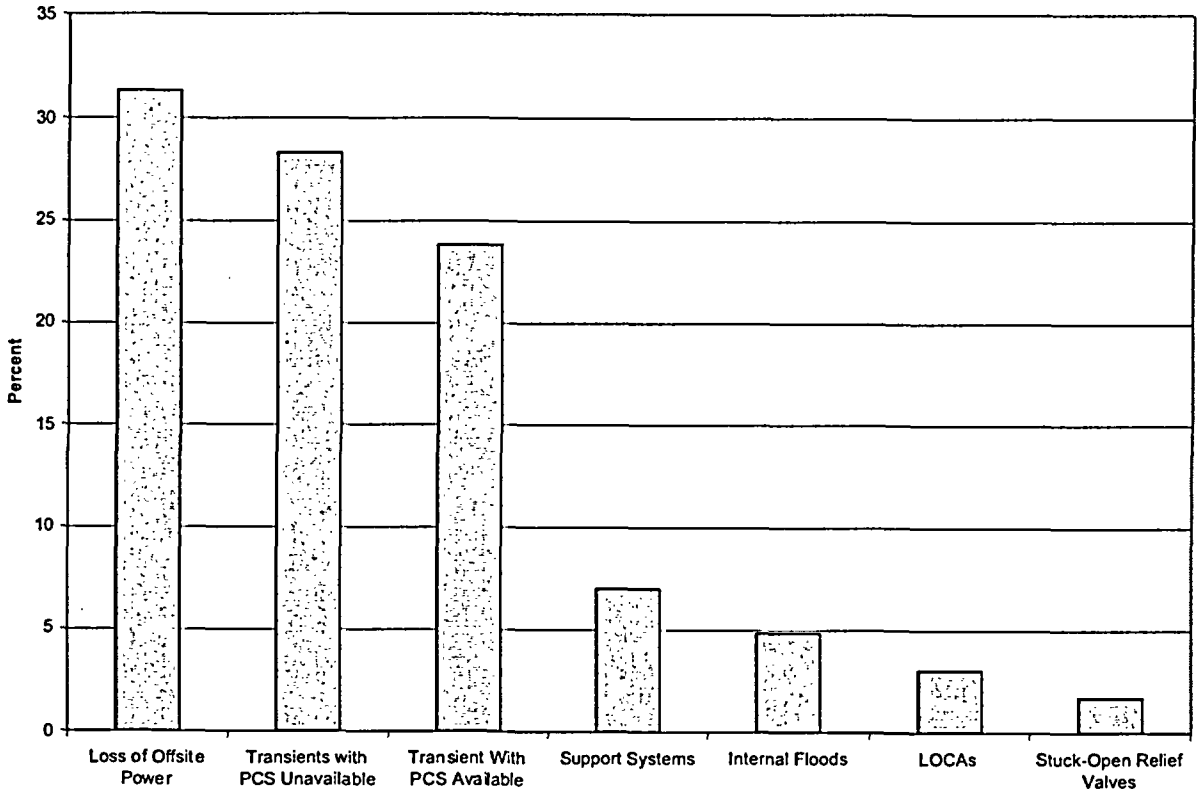

Figure 1-3
Browns Ferry Unit 3 Core Damage Frequency by Initiating Event Category

Table 1-2

Unit 3 Initiating Event Group Contributions to Core Damage Frequency

Initiating Event Category	Mean CDF (per reactor-year)	Percentage of Total
Loss of Offsite Power	1.05E-06	31.3
Transients with PCS Unavailable	9.53E-07	28.3
Transients with PCS Available	8.00E-07	23.8
Support System Failures	2.35E-07	7.0
Internal Floods	1.63E-07	4.8
LOCAs	1.01E-07	3.0
Stuck-Open Relief Valves	5.83E-08	1.7
Total	3.4E-06	99.9 (rounding)

Support system failure initiators (specifically, loss of plant air, loss of raw cooling water, or loss of either I&C bus 3A or 3B failures) contribute 7.0% to the total CDF.

Scenarios initiated by internal floods contribute 4.8% 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).

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 3.0% of the total CDF.

Scenarios initiated by the inadvertent opening of one or more safety relief valves (SRVs) contribute 1.7% to 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 which 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) which 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 1-2 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.

Table 1-3
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

Table 1-3
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	5
> 1E –08	35	28
>1E –09	378	59
> 1E –10	3179	83
> 1E –11	22,119	95
> 1E-12	22,119 + Not saved	100

The following presents a brief description of the 20 highest-ranked 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 second sequence includes additional failures (e.g., non-minimal).

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.

The fifth sequence is initiated by a total LOSP followed by the failure diesel generators 3A, 3B, 3C, and B. This combination of failures results in the 480V shutdown board 3B, which supplies room cooling to the B and D RHR pumps. Thus all Unit 3 RHR pumps are failed. The LPCI injection path is failed because of 3EB and 3EC diesel generators. HPCI fails long term because of the failure of diesel generator 3EA, which maintains the charger for long-term 250V DC power. RCIC fails long term because of its dependency on diesel generator A. Diesel Generator A and B are required to maintain the charging for long-term 250V DC power. Hence, there is no high-pressure injection. Suppression Pool cooling is failed due to electrical supports. Core damage occurs because of lack of injection

The sixth sequence is initiated by a total LOSP followed by the failure diesel generators 3A, 3B, 3C, and A. This combination of failures results in the loss of 480V shutdown board 3B, which supplies room cooling to the B and D RHR pumps. Thus all Unit 3 RHR pumps are failed. The LPCI injection path is failed because of 3EB and 3EC diesel generators. HPCI fails long term because of the failure of diesel generator 3EA, which maintains the charger for long-term 250V DC power. RCIC fails long term because of its dependency on diesel generator A. Diesel Generator A and B are required to maintain the charging for long-term 250V DC power. Hence, there is no high-pressure injection. Suppression Pool cooling is failed due to electrical supports. Core damage occurs because of lack of injection

The seventh sequence is also initialized by LOSP. Diesel generator 3A, 3B, and 3C fail. The RCIC pump also fails long term. Because of the dependency on shutdown board 3EA, HPCI fails in the long run. RHR pumps A, B, and C fail because of the failure of

diesel generators 3A, 3B, and 3C. RHR pump D fails because of its dependency on 480V shutdown board 3EB. The RHR low pressure injection path fails. Core damage occurs due to lack of injection.

Sequence eight is similar to sequence seven with the difference being the RCIC failure is in the injection phase.

A general transient initiates sequence nine. It is similar to sequence three but the failure of feedwater is caused by the failure to trip the turbine.

Sequence ten is a non-minimal version of sequence one. The additional failure is Unit 1/2 at power.

Sequence eleven is initiated by a loss of the 500kV to the unit. The loss causes feedwater to fail. HPCI and RCIC fail. Depressurization fails and core damage follows due to lack of inventory.

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

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

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

Sequence fifteen is also initialized by a total LOSP. This is followed by a failure of diesel generators B, C, 3EA, and 3EB. This combination of diesel generator failures causes a loss of EECW. Offsite power is not recovered prior to core damage.

Sequence sixteen is similar to sequence fifteen 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 seventeen is similar to sequence three, but has different failures in the containment event tree.

Sequence eighteen is similar to sequence two, but has different failures in the containment event tree.

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

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

Section Appendix A contains a listing of the top 50 core 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	1.57E-07	4.62E-02	4.62E-02
2	5.83E-08	1.71E-02	6.33E-02
3	5.76E-08	1.69E-02	8.03E-02
4	4.63E-08	1.36E-02	9.39E-02
5	4.38E-08	1.29E-02	1.07E-01
6	4.28E-08	1.26E-02	1.19E-01
7	3.85E-08	1.13E-02	1.31E-01
8	3.57E-08	1.05E-02	1.41E-01
9	2.95E-08	8.68E-03	1.50E-01
10	2.77E-08	8.15E-03	1.58E-01
11	2.55E-08	7.50E-03	1.66E-01
12	2.55E-08	7.50E-03	1.73E-01
13	2.43E-08	7.15E-03	1.80E-01
14	2.42E-08	7.12E-03	1.87E-01
15	2.40E-08	7.06E-03	1.94E-01
16	2.29E-08	6.74E-03	2.01E-01
17	2.13E-08	6.26E-03	2.07E-01
18	1.75E-08	5.15E-03	2.12E-01
19	1.67E-08	4.91E-03	2.17E-01
20	1.60E-08	4.71E-03	2.22E-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.

Table 1-5
Browns Ferry Unit 3 Important Operator Actions

Operator Action	Top Event/Split Fraction	Importance
Depressurize to Allow Low Pressure Injection	ORVD2	43.2
Open the Hardened Wetwell Vent	OLP2/3/4	9.2
Operator Aligns Suppression Pool Cooling	OSP	4.8
Align Alternate Injection to Reactor Vessel via the Unit 3 to Unit 2 RHR Crosstie*	U22	3.5
*The importance of the split fraction U22 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 3 Important Systems

System	% CDF
HPCI	55
RCIC	55
Diesel Generators	28
Feedwater/Condensate	13
Main Steam	10
RHR	9
RPS	7
Core Spray	3
RHRSW to RHR Loop II	2
Control Rod Drive	2
RHRSW to RHR Loop I	1
Standby Liquid Control	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 13% 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 3 falls in the mid-range for BWRs.

This release category group 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 3 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	Notes
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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1-4. 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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1-12. Deleted.

1-13. Deleted.

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APPENDIX A

UNIT 3 TOP RANKING SEQUENCES CONTRIBUTING TO CDF

1	LOCHS	222	1.5744E-007	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
2	LOCHS	226	5.8312E-008	SDRECF*OXF*DWF*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF*FC2*RBEF	NLERF
3	TRAN	632	5.7555E-008	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
4	ISLOCA	1	4.6342E-008		LERF
5	LOSP	5877	4.3780E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*GF4*EPR303*A3EAF*RXF*ROF*A3ECF*A3EBF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GB1*ABF*RSF*RHF*DKF*SDRECF*UB42CF*CBBF*UB43CF*RJ3F*RK3F*RL3F*DWF*RCWF*EAF*ECF*RBCF*SW2CF*SW1CF*PCAF*DCAF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*R480F*RPAF*RPCF*RPBF*RPDF*SPF*SPRF*LPCF*OAIF*NCDF*DWSF*RHSW3F*NOCDF	NLERF
6	LOSP	6199	4.2797E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*GF4*EPR303*A3EAF*RXF*ROF*A3ECF*A3EBF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GA1*AAF*RQF*REF*RMF*SDRECF*UB42CF*CBBF*UB43CF*RJ3F*DN3F*RK3F*RL3F*DWF*RCWF*EAF*ECF*RBCF*SW2AF*SW1AF*DCAF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*R480F*RPAF*RPCF*RPBF*RPDF*SPF*SPRF*LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF
7	LOSP	5332	3.8516E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*GF4*EPR303*A3EAF*RXF*ROF*A3ECF*A3EBF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*SDRECF*UB42CF*CBBF*UB43CF*RJ3F*RK3F*RL3F*DWF*RCWF*EAF*ECF*RBCF*DCAF*IVOF*RVC0*CDF*EPR63*RCL1*HPLF*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*R480F*RPAF*RPCF*RPBF*RPDF*SPF*SPRF*LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF
8	LOSP	5386	3.5735E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*GF4*EPR303*A3EAF*RXF*ROF*A3ECF*A3EBF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*SDRECF*UB42CF*CBBF*UB43CF*RJ3F*RK3F*RL3F*DWF*RCWF*EAF*ECF*RBCF*DCAF*IVOF*RVC0*CDF*RCI1*EPR63*HPLF*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*R480F*RPAF*RPCF*RPBF*RPDF*SPF*SPRF*LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF
9	TRAN	1016	2.9538E-008	SDRECF*OXF*DWF*TB1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
10	LOCHS	1245	2.7697E-008	SDRECF*OXF*DWF*U2AP1*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
11	L500U2	497	2.5487E-008	OG5F*EPR301*SDRECF*OXF*DWF*MCDF*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NL	NLERF

				ERFF*ELF*WWBF*WWF	
12	LOSP	6729	2.5480E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*GF4*GH4*EPR303*DGC1* A3EAF*RXF*ROF*A3ECF*A3EBF*RYF*RPF*A3EDF*UB41AF*UB41BF *UB42AF*UB42BF*SHUT1F*SHT2F*GAF*GDF*GBF*GCF*AAF*RQF*R EF*RMF*ABF*RSF*RHF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RI F*RJF*RNF*DLF*SDRECF*UB42CF*CBBF*UB43C F*RJ3F*DO3F*DN3F*RK3F*RL3F*DWF*RCWF*EAF*EBF*ECF*EDF*R BCF*SW2AF*SW1AF*SW2BF*SW1BF*SW2CF*SW1CF*SW2DF*SW1DF*P CAF*DCAF*CADF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF *HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*R480F*RPAF*RPCF*U2F*RP BF*RPDF*OSPF*LPCF*OAIF*NCDF*RHSW3F*NOC	NLERF
14	LOCHS	224	2.4334E-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*RBI2	NLERF
15	TRAN	262	2.4175E-008	SDRECF*OXF*DWF*BVR1*RVC0*FWHF*RCI1*HPI4*OBDF*ORVD2*FW AF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWB F*WWF	NLERF
16	LOSP	3970	2.4046E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GF2*EPR303*A3EAF*RXF*ROF *A3EBF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GB1*G C2*ABF*RSF*RHF*DKF*ACF*RRF*RFF*SDRECF*UB42CF*CBBF*UB4 3CF*DWF*RCWF*EAF*EBF*ECF*RBCF*SW2BF*SW2CF*SW1CF*PCAF* DCAF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPL F*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*RPAF*RPCF*U2F*RP BF*RPDF*OSPF*LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF
17	LOSP	3515	2.2936E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GH2*EPR303*A3EAF*RXF*ROF *A3EDF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GD1*G C2*ACF*RRF*RFF*ADF*RTF*RIF*RJF*RNF*DLF*SDRECF*UB42CF* CBBF*UB43CF*DWF*RCWF*EAF*EBF*EDF*RBCF*SW2BF*SW2DF*SW1 DF*PCAF*DCAF*OEEF*IVOF*RVC0*CDF*EPR63* RCLF*HPLF*FWAF*HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*RPAF*RPC F*U2F*RPBF*RPDF*OSPF*LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF
18	TRAN	636	2.1317E-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*FC2*RBEF	NLERF
19	LOCHS	223	1.7494E-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
20	FLTB2	168	1.6742E-008	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*ORVD2*FWAF*HRL F*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
21	LOAC	124	1.6020E-008	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*ORVD2*FWAF*HRL F*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*ELF*WWBF*WWF	NLERF
22	L500U2	38	1.5926E-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
23	LOPA	40	1.4261E-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
24	LRBCCW	83	1.4259E-008	SDRECF*OXF*DWF*RBCF*DCAF*IVOF*RVC0*FWHF*RCI1*HPI4*OIV	NLERF

				F*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLER FF*ELF*WWBF*WWF	
25	TRAN	433	1.3310E-008	SDRECF*OXF*DWF*MCD1*RVC0*FWHF*OBCF*FWAF*HSF*HXA1*HXC2 *U22*HXB5*HXD7*OSPF*OSDF*OLP2*NCDF*NOCDF	NLERF
26	LOCHS	244	1.2680E-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*OP3*IVR1*TRF*FC3*DWIF*RBE5	LERF
27	TRAN	1773	1.2149E-008	SDRECF*OXF*DWF*RXS1*OSL1*NAF*FWAF*HRLF*HR6F*SUFWF*CDA F*NCDF*NOCDF*NLERFF*CILF*IVR10*TR6*FCF*DWIF*RBE7	LERF
28	TRAN	1782	1.2149E-008	SDRECF*OXF*DWF*RXS1*OSL1*NAF*FWAF*HRLF*HR6F*SUFWF*CDA F*NCDF*NOCDF*NLERFF*CILF*WW1*IVR10*TR6*FCF*DWIF*RBE8	LERF
29	BOC	18	1.1795E-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
30	LOSP	4866	1.1567E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*EPR303*A3EAF*RXF*ROF *A3ECF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT 2F*GA1*AAF*RQF*REF*RMF*SDRECF*UB42CF*CBBF*UB43CF*RJ3F *DN3F*RK3F*RL3F*DWF*RCWF*EAF*RBCF*SW2AF*SW1AF*DCAF*IV OF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF* SUFWF*HSF*CDAF*CRDF*RPAF*RPBF*CS6*LPCF*OAIF*NCDF*NOCD	NLERF
31	TRAN	1020	1.0940E-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
32	LOSP	4618	1.0441E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*EPR303*A3EAF*RXF*ROF *A3ECF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT 2F*GB1*ABF*RSF*RHF*DKF*SDRECF*UB42CF*CBBF*UB43CF*RJ3F *RK3F*RL3F*DWF*RCWF*EAF*RBCF*SW2CF*SW1CF*PCAF*DCAF*IV OF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF* SUFWF*HSF*CDAF*CRDF*RPAF*HXCF*RPBF*CS6*LPCF*OAIF*NCDF *NOCDF	NLERF
33	LOSP	4483	1.0415E-008	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*EPR303*A3EAF*RXF*ROF *A3ECF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT 2F*SDRECF*UB42CF*CBBF*UB43CF*RJ3F*RK3F*RL3F*DWF*RCWF* EAF*RBCF*DCAF*IVOF*RVC0*CDF*EPR63*RCL1*HPLF*FWAF*HRLF *SUFWF*HSF*CDAF*CRDF*RPAF*RPBF*CS6*LPC F*OAIF*NCDF*NOCDF	NLERF
34	ISCRAM	201	1.0344E-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
35	LOCHS	1249	1.0258E-008	SDRECF*OXF*DWF*U2AP1*IVOF*RVC0*FWHF*RCI1*HPI4*OIVF*OR VD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*E LF*WWBF*WWF*FC2*RBEF	NLERF
36	FLTB2	104	1.0147E-008	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*FWAF*HRLF*HR6F *SUFWF*HSF*CDAF*LC1*JC1*NCDF*NOCDF*NLERFF*ELF*WWBF*WW F*IVR6	NLERF
37	TRAN	3078	1.0125E-008	SDRECF*OXF*DWF*U2AP1*MCD1*RVC0*FWHF*RCI1*HPI4*OBDF*OR VD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NLERFF*E LF*WWBF*WWF	NLERF

38	LOAC	60	9.7102E-009	SDRECF*OXF*DWF*MCDF*RVC0*CDF*RCI1*HPI4*FWAF*HRLF*HR6F *SUFWF*HSF*CDAF*LC1*JC1*NCDF*NOCDF*NLERFF*ELF*WWBF*WW F*IVR6	NLERF
39	LOSP	4502	9.6625E-009	OG5F*OG16F*UB43AF*UB43BF*GE1*GG2*EPR303*A3EAF*RXF*ROF *A3ECF*RYF*RPF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT 2F*SDRECF*UB42CF*CBBF*UB43CF*RJ3F*RK3F*RL3F*DWF*RCWF* EAF*RBCF*DCAF*IVOF*RVC0*CDF*RCI1*EPR63*HPLF*FWAF*HRLF *SUFWF*HSF*CDAF*CRDF*RPAF*RPBF*CS6*LPC F*OAIF*NCDF*NOCDF	NLERF
40	LOSP	2167	9.5933E-009	OG5F*OG16F*UB43AF*UB43BF*GF1*GH2*EPR303*A3EBF*A3EDF*U B41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GD1*GB2*GC4*A BF*RSF*RHF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RIF*RJF*RN F*DLF*SDRECF*UB42CF*CBBF*UB43CF*DWF*RCWF*EBF*ECF*EDF* RBCF*SW2BF*SW2CF*SW1CF*SW2DF*SW1DF*PCA F*DCAF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF*S UFWF*HSF*CDAF*CRDF*ORPF*RPAF*RPCF*U2F*RPBF*RPDF*OSPF* LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF
41	LRCW	247	9.5206E-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
42	LOSP	4194	9.4531E-009	OG5F*OG16F*UB43AF*UB43BF*GE1*GF2*EPR303*A3EAF*RXF*ROF *A3EBF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GA1*G B2*GC4*AAF*RQF*REF*RMF*ABF*RSF*RHF*DKF*ACF*RRF*RFF*SD RECF*UB42CF*CBBF*UB43CF*DN3F*DWF*RCWF*EAF*EBF*ECF*RBC F*SW2AF*SW1AF*SW2BF*SW2CF*SW1CF*PCAF*D CAF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF*HRLF*SUFW F*HSF*CDAF*CRDF*ORPF*RPAF*RPCF*U2F*RPBF*RPDF*OSPF*LPC F*OAIF*NCDF*RHSW3F*NOCDF	NLERF
43	L500U2	501	9.4398E-009	OG5F*EPR301*SDRECF*OXF*DWF*MCDF*RVC0*FWHF*RCI1*HPI4*O BDF*ORVD2*FWAF*HRLF*HR6F*SUFWF*HSF*CDAF*NCDF*NOCDF*NL ERFF*ELF*WWBF*WWF*FC2*RBEF	NLERF
44	LOSP	4128	9.4042E-009	OG5F*OG16F*UB43AF*UB43BF*GE1*GF2*EPR303*A3EAF*RXF*ROF *A3EBF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GD1*G B2*GC4*ABF*RSF*RHF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RI F*RJF*RNF*DLF*SDRECF*UB42CF*CBBF*UB43CF*DWF*RCWF*EAF* EBF*ECF*EDF*RBCF*SW2BF*SW2CF*SW1CF*SW2 DF*PCAF*DCAF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWAF* HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*RPAF*RPCF*U2F*RPBF*RPDF *OSPF*LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF
45	ELOCA	1	9.3900E-009	NCDF	LERF
50	LOSP	3671	9.1630E-009	OG5F*OG16F*UB43AF*UB43BF*GE1*GH2*EPR303*A3EAF*RXF*ROF *A3EDF*UB41AF*UB41BF*UB42AF*UB42BF*SHUT1F*SHT2F*GD1*G B2*GC4*ABF*RSF*RHF*DKF*ACF*RRF*RFF*ADF*RTF*RKF*RLF*RI F*RJF*RNF*DLF*SDRECF*UB42CF*CBBF*UB43CF*DWF*RCWF*EAF* EBF*EDF*RBCF*SW2BF*SW2CF*SW1CF*SW2DF*S W1DF*PCAF*DCAF*OEEF*IVOF*RVC0*CDF*EPR63*RCLF*HPLF*FWA F*HRLF*SUFWF*HSF*CDAF*CRDF*ORPF*RPAF*RPCF*U2F*RPBF*RP DF*OSPF*LPCF*OAIF*NCDF*RHSW3F*NOCDF	NLERF