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DIABLO CANYON INTERNAL EVENTS PRA REVIEW: METHODOLOGY AND FINDINGS*

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ABSTRACT

The review of the Diablo Canyon Probabilistic Risk Assessment (DCPRA) incorporated some new and innovative approaches. These were necessitated by the unprecedented size, scope and level of detail of the DCPRA, which was submitted to the NRC for licensing purposes. This paper outlines the elements of the internal events portion of the review citing selected findings to illustrate the various approaches employed. The paper also provides a description of the extensive and comprehensive importance analysis applied by BNL to the DCPRA model. Importance calculations included: top event/function level; individual split fractions; pair importances between frontline-support and support-support systems; system importance by initiator; and others. The paper concludes with a brief discussion of the effectiveness of the applied methodology.

1.0 BACKGROUND

The Diablo Canyon Probabilistic Risk Assessment[1] (DCPRA), presented by Pacific Gas and Electric (PG&E), represents an unprecedented PRA submittal to the NRC for direct licensing purposes. It is unprecedented in both size and level of detail and reflects the prime subcontractors' (Pickard, Lowe & Garrick - PLG) decade long experience in preparing more than 20 PRAs. The purpose of this paper is to outline the approach utilized in the review[2] of this work, to highlight some of the novel techniques applied and to comment on the overall success of the review approach.

The DCPRA was performed, at least in part, in response to a set of licensing conditions incorporated into the operating license for the Diablo Canyon plants. As such, it required formal review by the U.S. NRC. The Probabilistic Risk Assessment Branch, ORES was charged with the responsibility of the NRC review and Nilesh Chokshi was the NRC Program Manager. Brookhaven National Laboratory was contracted to conduct and integrate the detailed review. Within that scope,

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selected portions of the review were performed by others. The HRA methodology was reviewed by T. Ryan, NRC; the fire scenario portion of the review was conducted by A. Buslik, NRC; and the seismic review effort was headed up by M. Bohn of Sandia under contract to BNL. The focus of this paper, therefore, is mainly the internal events review performed directly by BNL.

The DCPRA quantified 50 initiating events grouped into six categories and screened out an additional seven categories. The fifty initiating events broke down as follows: nine LOCAs, fourteen Transients, six seismic levels, twelve fire/smoke scenarios and three flood/jet/spray scenarios. The modelling approach of the DCPRA was to create a number of modules and link them accordingly to develop a full spectrum of accident sequences. The modules broke down into three categories: 1) two support system event tree modules (one electrical and one mechanical), 2) seven early frontline event three modules and 3) four long term frontline event three modules. These were assembled as follows. First, the electrical event three with 21 top events was constructed, to each of the end points of the electrical event tree was attached the mechanical event tree with 13 top events. At this point, the support system model contained tens of thousands of end points. These were combined into "like" categories which resulted in 178 distinct support states. Each initiating event was then in turn solved for each support state with either early or long term frontline event tree modules attached (each appropriately modified to account for the given support states).

The above paragraph is meant to provide a brief outline of the size and level of detail of the DCPRA. Given the size and complexity of the DCPRA, it was determined that a novel approach would be required for the detailed review and analysis. The review itself was divided into two phases. The first phase was "interactive" and was conducted while the PRA was still being developed. The goal of this phase was to both familiarize the reviewers with the PRA and to provide a potential early feedback mechanism to the DCPRA team. During this initial phase, two site visits to the Diablo Canyon plant were made for familiarization purposes and three PRA workshops (approximately one-week each) were conducted. The second phase was to review of the final DCPRA report.

2.0 REVIEW APPROACH

The review strategy employed had to take into account the fact that neither the NRC nor the national laboratories had in-place processing software that could directly accommodate the DCPRA large event tree/small fault tree model. In addition, the strategy had to accept the fact that employing an independent requantification type of PRA review (with the use of the large fault tree/small event tree methodology) to a level of detail commensurate with that in the DCPRA, would simply be cost-prohibitive and unnecessary.

The resulting DCPRA review strategy, therefore, involved a <u>detailed review</u> of <u>selected</u> portions of <u>each</u> of the major elements of the DPCRA. As the actual review progressed, some elements received more attention than others according to the perceived needs by the reviewers. The following seven point plan was developed by BNL as the overall review basis for the DCPRA:

represented an expression that combined all the failure modes of each of the elements of the supercomponents. BNL also checked the equation against the plant drawings, test/maintenance procedures, and Technical Specifications to verify that all major components/failure modes/unavailabilities were included.

In order to then verify the various split fractions associated with each fault tree, BNL had to set various elements to one or zero to define each boundary condition and then solve that version of the fault tree four times to account for the different postulated sets of system alignment. The methodology of systems analysis applied in the DCPRA requires that the top event split fraction (associated with a system under a given boundary condition) should reflect the notion that the system (or its portion) in question is in one of the following mutually exclusive alignments: 1) normal alignment, 2) testing alignment, 3) maintenance alignment, or 4) misalignment. Thus, the contribution to the system unavailability from a specific alignment is determined by the conditional system unavailability, given that the system is in that alignment multiplied by the fraction of time that the system spends in that alignment. The quantification/verification of the conditional split fractions in most cases provided good agreement with the PG&E results. The difference in the majority of the cases coming from some modeling errors of minor significance and from the use of Monte Carlo techniques by PG&E and point estimates by BNL.

The following systems/functions were subjected to detailed review/requantification:

- High Pressure Injection Function
- Low Pressure Injection Function
- Auxiliary Feedwater System
- Diesel Generator & Diesel Fuel Transfer Systems
- Electrical Power Systems (AC & DC)
- Auxiliary Saltwater System
- Component Cooling Water System
- Solid State Protection/Reactor Protection Systems

2.3 DATA ANALYSIS

BNL carried out the following types of analyses to verify the DCPRA data base. The DCPRA data base was derived from the PLG proprietary data base and updated using Bayesian techniques to incorporate Diablo Canyon - specific data/experience. As part of the auxiliary feedwater system review, BNL solved the derived fault trees with first the DCPRA data and then with an alternate generic data base derived from other recent PRAs. This was done to see the sensitivity of the model to the different data bases. The quantification of the conditional split fractions was in fairly close agreement; demonstrating little sensitivity to the two data bases. Had the data bases provided significantly divergent results, further review effort would have been devoted to this particular area of the review.

In terms of initiating event quantification, BNL checked all of the initiators against other industry sources. A number of the initiating event frequencies seemed somewhat low. This was attributed to the rather restrictive criteria applied by PG&E to select some prior event samples for Bayesian updating (mainly transients). However, use of less restrictive selection criteria in sensitivity

1. The logic for the primary event trees will be reviewed to verify consistency and accuracy.
2. Selected frontline and support systems will undergo an independent fault tree analysis to verify the DCPRA's approach to unavailability modelling (the systems will be selected based upon perceived importance). This effort will include requantification of an appropriate number of top event conditional split fractions.
3. Selected failure probabilities and initiating event frequencies will be reviewed (including the Bayesian updating process) to verify the DCPRA data analysis. Actual failure data selection will be determined by the results of item 1 above.
4. An abbreviated fault model of the entire Diablo Canyon plant will be developed by incorporating the leading accident sequences from the DCPRA.
5. Given the fault model from item 4 above, investigation will be undertaken on the impact of the findings from items 1 through 3 above as well as the performance of other analyses such as importance measures, pair-importance, and sensitivity calculations.
6. In addition to the above overall review plan, two novel aspects of the DCPRA which are a) the approach to human reliability analysis and b) the relay chatter analysis will receive special attention.
7. The seismic portion of the PRA review will follow a similar overall methodological approach modified as necessary to account for the specifics of the seismic analysis.

2.1 EVENT TREES

In terms of item 1 above, the entire set of DCPRA event trees was not given a rigorously detailed review by BNL as part of the overall review process. The basis for this was that there was an extremely detailed and comprehensive methodology applied to the event tree development and, therefore, BNL believed that the review effort should concentrate resources on other areas of the PRA. The DCPRA methodology utilized event sequence diagrams (ESDs) and stressed the involvement of both PRA analysts and plant operations personnel. BNL did check for any obvious errors/omissions in the event tree structures but none were apparent.

2.2 SYSTEMS ANALYSIS

The fault tree analysis portion of the review was conducted as part of the systems analyses. The system documentation associated with the DCPRA provided reliability block diagrams (as opposed to actual fault trees) containing supercomponents covering large portions of the system. BNL converted these diagrams into fault trees and used the SETS[3] computer code to solve them. This allowed BNL to display the leading cut sets for those top events so modelled. Such cut sets are not provided within the DCPRA. In addition, the fault trees had to be prepared according to the specific requirements of the α - factor common cause failure methodology.

The quantification of the supercomponents was supplied in algebraic equation form by PG&E. That is, in order for BNL to supply the value block for input to the SETS code, the algebraic equation for each of the supercomponents had to be computed as well as broken down to identify its constituent parts. Each equation

Birnbaum importances for each initiator. As it may also be of specific interest as to how many leading sequences contribute to the core damage frequency for a given initiator, this information is given in the last column of Table 1.

In order to gain insights into the vulnerability of the Diablo Canyon plant with respect to system-level failures, a system-level importance analysis was performed. The analysis was separately carried out for support systems and frontline systems as well as for operator and recovery action failures explicitly appearing in the event sequences as top event split fractions. The analysis was global, in that sense, that it did not distinguish between the various initiating events. In the analyses each system/operator action importance was determined by calculating the importance of its associated top event or an aggregate of top events appearing in the DSM.

Table 2 presents the leading unnormalized Fussel-Vesely importances for both the overall systems/safety functions as well as their constituent top events. Part A lists the support system, Part B the frontline system and Part C the operator/recovery action importances. The most important support systems are: 1) the diesel generator systems, and 2) Unit 1 125 V DC power. The most important frontline systems are: 1) the auxiliary feedwater and 2) the primary pressure relief systems. The most important operator action is to maintain hot-standby given a transient.

In order to gain insights into the importances of the individual top event split fractions, BNL performed a dedicated top event split fraction importance analysis. A complete list of top event split fractions ranked according to their Birnbaum importance is given in an Appendix of the final review report[2].
The overall ranking of the leading top event conditional split fraction (CSF) importances is as follows:

		Leading CSFs	Normalized* Fuss-Ves. Importances (%)
1.	Operator inability to maintain hot standby everything available).	HSI	11.5
2.	Loss of primary pressure relief (loss of PORV operability for feed and bleed. No instrument air.)	OBI	9.9
3.	Primary pressure relief (for LOOP/SGTR, failure of 1/2 PORVs or 1/3 SRVs).	PRD	9.1
4	Loss of DG13 (after loss of 4.15kV bus HF).	GF1	8.6
5.	Failure to trip RCP after loss of CCW system to prevent seal LOCA	RP2	6.8
6.	Loss of DG12 (DG13 is successful).	GG1	6.3

*For normalization, the total non-seismic core damage frequency was used.

studies did not result in large variations in total core damage frequency. Additionally, BNL selected two initiators for detailed scrutiny. The loss of auxiliary saltwater (LOSW) and the loss of component cooling (LPCC) were selected for this purpose. Both of these initiators were quantified by fault tree analyses in the DCPRA and the latter initiator was basically limited to loss of the CCW pumps (thus LPCC rather than LOCC). BNL's approach was to carry out a detailed industry-wide LER-type search for all LOSW and LOCC events. BNL then screened this list for events that, due to design considerations, could not happen at Diablo Canyon and then proceeded to undertake a Bayesian updating of this data with the Diablo Canyon experience, (i.e. no events in either category). This effort yielded significantly larger initiating frequencies and, therefore, significantly large core damage contributions from these two initiators than that presented in the DCPRA. Following meetings with the DCPRA team (Pacific Gas and Electric, et. al.), PG&E submitted new and higher values for both LOSW and LPCC. The increases were 44 percent and 47 percent respectively.

2.4 DOMINANT SEQUENCE EVALUATION

The abbreviated fault tree model was originally going to be developed by BNL, however, PG&E developed a reduced model (Dominant Sequence Model - DSM) for their own purposes and agreed to share this with BNL. The PGE model contained both internal and the non-seismic external events and therefore the BNL results based upon this model were termed "non-seismic" results. The leading sequences and the quantification associated with all of the conditional split fractions and basic event failure probabilities were provided to BNL on a floppy disk. BNL had to modify the model into a Boolean expression and then utilized this model as the basis for the quantification described in Section 3.

3.0 IN-DEPTH IMPORTANCE ANALYSES

Initial documentation of the DCPRA and its results was limited to Chapter 6 of the Long Term Seismic Program (LTSP) Final Report. As such, a significant amount of information required for the review as well as insights that might be derived from the PRA were missing. The review process subsequently surfaced considerably more information and, because of the initial paucity of documented insights, also sought to independently offer insights where feasible. To this end, BNL performed detailed initiator, system-level/safety function and top event importance analyses based on the DSM. The results of a sampling of these review efforts are presented herein to illustrate the scope, depth, and novel approaches employed.

3.1 OVERALL IMPORTANCE MEASURES

Based on the DSM, the core damage frequency contributions for the non-seismic initiating events were calculated. Table 1 lists the ranked Fussel-Vesely importances (unnormalized and normalized) of the initiating events included in the DSM. In order to gain insights into the plant non-mitigation probability given the occurrence of an initiating event, another quantity: the conditional core damage probability was also calculated for each initiating event. This quantity is also known as the Birnbaum importance. The Birnbaum importance has the advantage that it is independent of the initiator frequency itself (which may change significantly) but actually measures the plant performance under the condition of the occurrence of that initiating event. Table 1 also shows the concepts. They are determined by calculating the importances of the intersection between two aggregates of top event split fractions, where each aggregate contains the top event split fractions associated with a given system or function. The unnormalized Fussel-Vesely importances of support system pairs, as well as those of frontline system-support system pairs are tabulated in matrix form in Tables 4 and 5 respectively.

From Table 4, the overall ranking of the top five support system-support system pair importances is as follows:

1. Component Cooling Water - Diesel Generator Systems
2. Component Cooling Water - Vital 125V DC Systems
3. Diesel Generator Systems - Vital 125V DC Systems
4. Control Room Ventilation - Diesel Generator Systems
5. 480V Switchgear Ventilation - Diesel Generator Systems

From Table 5, the overall ranking of the top five frontline system-support system pair importances is as follows:

1. Primary RCS Pressure Relief - Diesel Generator Systems
2. Auxiliary Feedwater System - Diesel Generator Systems
3. Primary RCS Pressure Relief - Instrument AC Power
4. Auxiliary Feedwater System - Instrument AC Power
5. Auxiliary Feedwater System - Vital 125V DC

The pair importances presented herein reflect aggregated split fractions and in some cases aggregated top events to represent the system/function level. Unnormalized Fussel-Vesely importances as well as the associated Birnbaum importances were also calculated for a variety of combinations of all top event individual split fractions of the DSM. These are listed in ranked form (according to the unnormalized Fussel-Vesely importance) in nine tables within Appendix D2 of the final report[2]. Each of those tables provides some additional insight into the plant safety.

4.0 FINDINGS WITH RESPECT TO THE REVIEW PROCESS

There were two primary goals associated with the review process. The first was to ensure that the DCPRA was sufficiently complete and accurate to provide a reasonable foundation upon which the necessary elements of the Diablo Canyon Long Term Seismic Program (LTSP) could be based. The second was to provide quality feedback, where appropriate, so that the DCPRA might become an even more useful tool in any future applications.

We believe that both goals were met in that the review was sufficiently rigorous and broad enough in scope for us to conclude with a high degree of confidence that the DCPRA does indeed provide a reasonable foundation to support the LTSP and sufficient feedback was provided such that some elements of the DCPRA were modified during the review and others have been identified by PG&E for future revisions.

One of the key elements of the review process turned out to be its interactive nature. As discussed previously, the first phase of the review was termed the interactive phase, however, the formal review turned out to be even more

3.2 SYSTEM/TOP EVENT IMPORTANCES BY INDIVIDUAL INITIATIOR

A PRA and/or its review should be able to give quantitative answers to questions posed frequently in connection with nuclear plant safety. For example: "Given an initiating event of a certain type, which are those safety systems/operator actions whose unavailability/failure probability dominate the failure to mitigate the variety of event scenarios that may follow that initiator?" or: "Given a safety system or operator action with its characteristic unavailability/failure probability, which are those initiating events where this contributes most to the core damage frequency?"

In order to supply these answers for the Diablo Canyon plant, BNL extended its system's importance analysis to individual initiating events. An analysis was performed for each of the initiating events of the Dominant Sequence Model. Tables 3 presents the results of these analyses for the internal event initiators. For each initiator, the unnormalized Fussel-Vesely importances of system/operator actions and associated top events were calculated.

In terms of Table 3, when one scans the data column below a given initiator (put sheets 1, 2, or 3, 4 together vertically) one can read off the answer to a question of the first type above. When one scans the data row belonging to a system/operator action (put sheets 1, 3, or 2, 4 together horizontally) one can get an answer to a question of the second type. For example, given the initiating event RT; the ranking of system/operator action importances is:

1. Auxiliary Feedwater System
2. Maintain Control for Hot Standby
3. Primary RCS Pressure Relief (feed and bleed)
4. Instrument AC Power, etc.

Or, given the Auxiliary Saltwater System, the ranking of the initiating event importances is:

1. Loss of One 125V DC Bus, L1DC
2. Loss of Offsite Power, LOOP

3.3 PAIR IMPORTANCES.

Individual system/top event split fraction pair importances provide information that can be used to identify system/system and system/human action unavailabilities, whose simultaneous occurrence are critical with regard to the core damage frequency. The identification of these pairs is therefore relevant to plant safety from an operational point of view; it guides the personnel, e.g., to assess the advisability of permitting simultaneous activities (maintenances, tests) on two systems that may not be otherwise prohibited by the Technical Specifications.

Pair importance characterizes the contribution of the intersection of the pair (split fractions) to the total core damage frequency. To obtain normalized pair-wise Fussel-Vesely importances, the above quantities should be divided by the normalization constant; in this case the total non-seismic core damage frequency.

The pair-wise system-level importances represent a generalization of the above

Table 1
Initiating Event Contributions to Non-Seismic Core Damage Frequency
Dominant Sequence Model

Initiating Event				Importance			
No. (1)	Designator	Category	Frequency, Ii (Per Year)	Unnormalized FUSS-VES	FUSS-VES (%)	BIRNBAUM	# of CD Sequences
1	LOOP	Loss of Offsite Power	9.10-02	4.18-05	23.57	4.59-04	183
2	CRFIRE *	Control Room and Cable Spreading Room Fires		3.17-05	17.87		1
3	RT	Reactor Trip	1.14+00	1.62-05	9.13	1.42-05	34
4	TT	Turbine Trip	1.05+00	1.48-05	8.34	1.41-05	33
5	PLMFW	Partial Loss of Main Feedwater	7.49-01	1.08-05	6.09	1.45-05	26
6	L1DC	Loss of One DC Bus	2.56-02	9.50-06	5.36	3.71-04	34
7	FS8	Fire Scenario: Loss of 4.16kV Buses HF, HG and HH	6.48-06	6.48-06	3.65	1.00+00	2
8	FS11	Flood Scenario: Loss of Auxiliary Saltwater	3.81-04	6.20-06	3.50	1.63-02	4
9	MLOCA	Medium LOCA	4.63-04	5.97-06	3.37	1.29-02	7
10	SGTR	Steam Generator Tube Rupture	1.71-02	3.58-06	2.22	2.10-04	12
11	LPCC	Total Loss of Component Cooling Water	1.96-04	3.19-06	1.80	1.63-02	4
12	EXFW	Excessive Feedwater Flow	2.79-01	3.12-06	1.76	1.12-05	9
13	SLBO	Steam Line Break Outside Containment	5.53-03	2.80-06	1.58	5.06-04	24
14	LLOCA	Large LOCA	2.02-04	2.58-06	1.45	1.28-02	4
15	SLBI	Steam Line Break Inside Containment	4.63-04	2.38-06	1.34	5.15-03	8
16	SLOCI	Small LOCA: Isolable	1.61-02	1.81-06	1.02	1.12-04	6
17	LOSWV	Loss of 480V Switchgear Ventilation	6.29-05	1.61-06	.91	2.56-02	6
18	FS1	Fire Scenario: Loss of Both Motor-Driven AFW Pumps	2.94-04	1.47-06	.83	5.00-03	9
19	LOSW	Total Loss of Auxiliary Saltwater	9.74-05	1.45-06	.82	1.49-02	2
20	LOCV	Loss of Control Room Ventilation	7.99-02	1.24-06	.70	1.55-05	6
21	FS6	Fire Scenario: Loss of 4.16kV Buses HF and HG	2.42-05	1.10-06	.62	4.54-02	2
22	LOPF	Loss of Primary Flow	1.21-04	1.08-06	.61	8.89-06	5
23	1MSIV	Closure of One MSIV	1.07-01	9.51-07	.53	8.89-06	5
24	TLMFW	Total Loss of Main Feedwater	9.98-02	8.87-07	.50	8.89-06	5
25	SLOCN	Small LOCA: Non-Isolable	5.26-03	8.17-07	.46	1.55-04	4
26	LCV	Loss of Condenser Vacuum	8.73-02	7.76-07	.44	8.89-06	5
27	FS9	Flood Scenario: Loss of All AFW	1.35-05	6.87-07	.39	5.09-02	2
28	ISI	Inadvertent Safety Injection Signal	7.39-02	6.57-07	.37	8.89-06	5
29	FS5	Fire Scenario: Loss of Auxiliary Saltwater	5.26-05	5.71-07	.32	1.09-02	1
30	VSI(SS)	Interfacing LOCA (RHR Suction Side)	1.01-06	5.00-07	.28	4.95-01	1
31	HAZCHM	Chemical Hazard (e.g., chlorine/ammonia releases)	4.39-04	3.51-07	.20	7.99-04	1
32	ELOCA	Excessive LOCA	2.66-07	2.66-07	.15	1.00+00	1
33	FS10	Flood Scenario: Loss of Both Motor-Driven AFW Pumps	1.40-05	2.93-08	.02	2.10-03	1
		Total Internal		1.29-04	72.72		
		Total "External"		4.84-05	27.28		
		Total CDF (Dominant Sequence Model)		1.77-04			452

*Sum of six control room and cable spreading room fire sequences which break down as follows:

CR-VB-1	CR Vertical Board-1: Loss of ASW, CCW controls	1.08-04	1.25-06	.70	1.16-02
CR-VB-2	CR Vertical Board-2: Loss of PORV and Charging Pump controls	8.00-05	1.16-06	.65	1.45-02
CR-VB-2/3	CR Vertical Boards 2 and 3, Interface: Loss of PORV and AFW controls	9.36-05	3.15-06	1.76	3.37-02
CR-VB-4	CR Vertical Board 4: Loss of 4.16kV Buses HF,HG & HH	9.74-05	6.01-06	3.38	6.17-02
CSR-1	Cable Spreading Room: Loss of ASW, CCW controls	5.49-04	7.90-06	4.45	1.43-02
CSR-2	Cable Spreading Room: Loss of PORV and Pressurizer Instrumentation	9.25-04	1.23-05	6.93	1.33-02

interactive. All eight system analysis reviews listed in Section 2.2 were documented as they were accomplished in letter reports to the NRC Program Manager. These reports were forwarded to PG&E and meetings were held to discuss the preliminary findings. Each meeting covered two to three letter reports.

As with any large and complex piece of work such as the DCPRA, it is almost impossible to document every detail, assumption, success criterion, etc. Therefore, when the meetings were held, much of the open item material was found to be because of insufficient documentation. Other opeb items were shown to have merit with some being dismissed as having very low impact and others accepted in whole or in part as feedback into the DCPRA.

Finally, we believe that the rather sophisticated importance analyses carried out by BNL provided a large number of insights with respect to the Diablo Canyon plant that were not otherwise available.

REFERENCES

1. Long Term Seismic Program Final Report, submitted to NRC by PG&E Letter No. DCL-88-192, July 31, 1988 (Chapter Six: Probabilistic Risk Analysis).

2. BNL Review of the Diablo Canyon Probabilistic Risk Analysis, to be published as a NUREG/CR.

3. SETS Reference Manual, NUREG/CR-4213, May 1985.

Table 2 (Continued) (Sheet 2 of 3)

B Reduced Model, Frontline Systems

Frontline System	Associated Top Event(s)	Fussel-Vesely Importances: Top Event Importance	Unnormalized System Importance	Syst. Imp. (%)
Auxiliary Feedwater System			4.586-05	25.9
	AW	4.586-05		
	TD	---		
Primary RCS Pressure Relief			3.717-05	21.0
	PR	1.689-05		
	PO	---		
	OB	2.028-05		
ECCS, Low Pressure			1.390-05	7.8
	LA	7.519-06		
	LB	7.149-06		
	LV	2.125-07		
	RW	2.072-07		
	VA	2.292-07		
	VB	7.663-07		
	AC	1.267-06		
	LI	---		
	MU	1.918-06		
ECCS, High Pressure			7.456-06	4.2
	CH	8.943-07		
	SI	7.268-07		
	HR	1.085-06		
	RC	---		
	RF(-RF4)	4.794-06		
Reactor Vessel Integrity After Pressurized Thermal Shock (PTS)	VI	7.175-06	7.175-06	4.0
Turbine Trip and Main Steam Isolation			5.984-06	3.4
	TT	---		
	MS	5.984-06		
Isolation of Ruptured SG	SL	1.940-06	1.940-06	1.1
Interfacing LOCA Tree Top Events			5.0-07	.3
	VO,VC,VR,SM	---		
	IT	5.0-07		
	LW	---		
	ME	5.0-07		

Table 2
System/Operator Action Importances for Non-Seismic Core Damage Frequency
Ranking According to System/Operator Action Importances
(Sheet 1 of 3)

A. <u>Dominant Sequence Model, Support Systems</u>

Support System		Fussel-Vesely Importances		
		Unnormalized		
	Associated Top Event(s)	Top Event Importance	System Importance	Syst. Imp (%)
Diesel Generator Systems			4.255-05	24.0
a. Unit 1 DGs	GF	1.517-05		
	GG	1.983-05		
	GH	2.139-05		
b. Unit 2 DGs	TG	7.387-06		
	TH	7.099-06		
c. Swing Diesel Alignment	SW	9.262-06		
d. Diesel Fuel Oil Transfer	FO	7.004-06		
Vital 125V DC Power, Unit 1			1.681-05	9.5
	DF	2.281-06		
	DG	3.926-06		
	DH	1.006-05		
Instrument AC Power			1.138-05	6.4
	I1	3.675-06		
	I2	1.771-06		
	I3	4.159-06		
	I4	1.771-06		
Component Cooling Water	CC	1.065-05	1.065-05	6.0
Vital AC Power, Unit 1			8.605-06	4.9
	AF	2.428-06		
	AG	6.722-07		
	AH	5.500-06		
	SF,SG,SH	---		
Solid State Protection System			5.153-06	2.9
	SA	4.000-05		
	SB	4.376-05		
480V Switchgear Ventilation	SV	4.411-06	4.411-06	2.5
Auxiliary Saltwater	AS	2.588-06	2.588-06	1.5
Control Room Ventilation	CV	2.583-06	2.583-06	1.5
Reactor Protection System	RT	1.558-06	1.558-06	0.9

Table 3
Unnormalized System/Operator Action Importances for Internal Event Initiators
Dominant Sequence Model (Sheet 1 of 4)

System/Operator Action	Associated Top Events Or Their Total	Initiator, Initiator Frequency (yr⁻¹) LOOP IF=9.10-02	RT,IF=1.14 TT,IF=1.05	PLMFW IF=7.49-01	LIDC IF=2.56-02	MLOCA IF=4.63-04	SGTR IF=1.71-02	LPCC,LOSW IF=1.96 04 IF 9.74-05	EXFW IF 2.79-01	SLBO IF 5 53
Support Systems										
Non-Vital Electric Power	OG	Initiator	5.762-07	2.927-07	---	---	---	---	---	---
Diesel Generator System	Total	4.532-04	5.762-07	2.927-07		---	---	---	---	
a. Unit 1 DGs	GF	1.544-04	4.120-07	2.927-07	---	---	---	---	---	---
	GG	2.100-04	2.773-07	1.579-07	---	---	---	---	---	---
	GH	2.257-04	2.926-07	2.926-07	---	---	---	---	---	---
b. Unit 2 DGs	TG	7.839 05	1.194 07	---	---	---	---	---	---	---
	TH	7.800-05	---	---	---	---	---	---	---	---
c. Swing Diesel Align.	SW	1.018-04	---	---	---	---	---	---	---	---
d. Diesel Fuel Oil Transfer	FO	7.491-05	1.641-07	---	---	---	---	---	---	---
Instrument AC Power	Total	3.804-06	3.292-06	3.292-06	5.638-07	---	2.251-05	---	---	1.504-04
	I1	---	1.097-06	1.097-06	---	---	7.498-06	---	---	5 854 05
	I2	---	5.943-07	9.915-07	---	---	3.756-06	---	---	1.665 05
	I3	3.804-06	1.097-06	1.097-06	5.638-07	---	7.498-06	---	---	5 854 05
	I4	---	5.493-07	5.493-07	---	---	3.756-06	---	---	1.665 05
Auxiliary Saltwater	AS	1.702-05	---	---	2.476-05	---	---	Initiator	---	---
Vital 125V DC Power, Unit 1	Total	3.062-05	1.945-06	1.852-06	1.682-04	---	1.376-05	1.406-03	1.692-06	1 049-04
	DF	7.815-06	1.520-07	1.520-07	5.750-05	---	4.597-06	---	---	2.037 05
	DG	9.809-06	4.422-07	4.422-07	---	---	4.597-06	7.050-04	4.307-07	4 210 05
	DH	1.299-05	1.351-06	1.258-06	1.107-04	---	4.571-06	7.010-04	1.261-06	4.246 05
Component Cooling Water	CC	4.374-05	1.015-06	1.015-06	9.635-05	---	---	---	8 113-07	---
Vital AC Power, Unit 1	Total	1.825-05	2.737-07	9.827-07	1.661-04	---	---	---	8.581-07	6 192 05
	AF	5.566-06	1.246-07	1.246-07	5.644-05	---	---	---	---	2.000 05
	AG	4.944-06	---	---	---	---	---	---	---	---
	AH	7.737-06	1.491-07	8.581-07	1.097-04	---	---	---	8.581-07	4.192-05
Control Room Ventilation	CV	2.452-05	---	---	--	7.699-04	---	---	---	---
Solid State Protection System	Total	4.056-06	7.149-07	7.149-07	9.719-06	9.620-04	---	---	---	---
	SA	---	7.149-07	7.149-07	4.434-06	9.620-04	---	---	---	---
	SB	4.056-06	7.149-07	7.149-07	5.285-06	9.620-04	---	---	---	---
480V Switchgear Ventilation	SV	1.752-05	5.363-07	2.575-07	---	---	---	---	1 710-06	---
Reactor Protection System	RT	---	3.511-07	3.511-07	---	---	---	---	---	---
Vital AC and DC Power, Unit 2	Total	8.254-06	---	---	---	---	---	---	---	---
	BF	---	---	---	---	---	---	---	---	---
	BG	5.937-06	---	---	---	---	---	---	---	---
	BH	2.318-06	---	---	---	---	---	---	---	---
Frontline Systems										
Auxiliary Feedwater System	AW	1.602-04	6.470-06	7.334-06	2.498-04	---	---	---	3 637-06	1 909 04
Primary RCS Pressure Relief	Total	1.839-04	4.416-06	4.454-06	1.097-04	---	---	---	1 078 06	1 823 04
	PR	1.745 04	---	---	4.815-06	---	---	---	- -	-
	OB	9.436-06	4.416-06	4.454-06	1.049-04	---	---	---	1 078 06	1 823 04
ECCS, Low Pressure	Total	5.090 05	1.948-07	1.773-07	1.738-06	4 140-03	1.334-04	---		1 146 04
	LA	2.284-05	1.948-07	1.773-07	---	3 681-03	4.981 05	---		1 146 04

Table 2 (Continued) (Sheet 3 of 3)

C. Reduced Model, Operator and Recovery Actions

Operator Action	Associated Top Event(s)	Fussel-Vesely Importances: Unnormalized Top Event Importance	Fussel-Vesely Importances: Unnormalized Op. Action Importance	Operator Action Imp. (%)
Maintain Control for Hot-Standby After an Accident	HS	1.960-05	1.960-05	11.0
Operator Trips RCPs After Loss of CCW to Prevent Seal LOCA	RP	1.215-05	1.215-05	6.8
Actions Needed to Maintain RCP Seal Cooling	SE	8.999-06	8.999-06	5.1
Electric Power Recovery Factors			5.958-06	3.4
	RESLC1	1.645-06		
	RESLC2	1.484-06		
	RESLC3	9.360-08		
	REAC06	2.733-06		
	REAC12	2.925-09		
Secure SI Per Operating Procedures Following SGTR	OP	1.643-06	1.643-06	0.9
Various Human Failures in Accident Recovery			1.136-06	0.6
	ZHESV3	2.874-07		
	ZHEHS5	3.508-07		
	ZHEAW4	8.748-08		
	ZHERP2	1.709-07		
	ZHESW1	2.236-08		
	ZHERE2	2.018-08		
	ZHEAW3	2.584-08		
	ZHEF06	1.153-07		
	ZHEOB2	5.587-08		
Operator Actuation of SSPS Signal	OS	1.069-06	1.069-06	0.6

System/Operator Action	Associated Top Events Or Their Total	Initiator, Initiator Frequency (yr^{-1}) LOOP IF=9.10-02	RT,IF=1.14 TT,IF=1.05	PLMFW IF=7.49-01	LIDC IF=2.56-02	MLOCA IF=4.63-04	SGTR IF=1.71-02	LPCC,LOSW IF=1.96-04 IF=9.74-05	EXFW IF=2.79-01	SLBO IF=5.53-03
	LB	2.054-05	1.773-07	1.773-07	9.166-08	3.681-03	4.981-05	---	---	1.146-04
	LV	---	---	---	---	4.590-04	---	---	--	---
	RW	---	---	---	---	---	---	---	---	---
	VA	2.519-06	---	---	---	---	---	---	---	---
	VB	7.958-06	---	---	1.646-06	---	---	---	---	---
	AC	---	---	---	---	---	---	---	---	---
	MU	---	---	---	7.733-08	---	8.363-05	---	---	---
ECCS, High Pressure	Total	4.007-05	1.481-08	---	2.732-06	5.022-03	---	---	---	---
	CH	9.065-06	1.210-08	---	1.371-08	9.165-05	---	---	---	---
	SI	5.295-06	---	---	3.949-07	9.165-05	---	---	---	---
	HR	1.155-05	---	---	1.299-06	---	---	---	---	---
*(-RF4)	RF*	1.416-05	2.712-09	---	1.024-06	4.930-03	---	---	---	---
Reactor Vessel Integrity	VI	1.364-05	8.206-07	8.206-07	7.518-06	2.000-03	3.978-05	---	8.206-07	8.291-05
Turbine Trip & Main Steam Isolation	MS	1.772-06	3.638-07	3.638-07	5.750-05	---	---	---	---	5.063-04
Isolation of Ruptured SG	SL	---	---	---	---	---	1.134-04	---	---	---
Containment Isolation	CI	1.023-06	---	---	7.011-07	---	---	---	---	---
Containment Spray	Total	---	---	---	1.165-06	---	---	---	---	---
	CS	---	---	---	1.314-08	---	---	---	---	---
	SR	---	---	---	1.152-06	---	---	--	---	---
Operator/Recovery Actions										
Maintain Control for Hot Standby After an Accident	HS	7.515-06	4.991-06	4.991-06	---	---	---	---	5.010-06	---
Operator Trips RCPs After Loss of CCW to Prevent Seal LOCA	RP	---	7.803-07	4.968-07	---	---	---	1.086-02	---	---
Actions Needed to Maintain RCP Seal Cooling	SE	5.564-06	2.041-07	2.041-07	---	---	---	1.086-02	---	---
Electric Power Recovery Factors	Total	6.548-05	---	---	---	---	---	---	---	---
	RESLC1	1.808-05	---	---	---	---	---	---	---	---
	RESLC2	1.631-05	---	---	---	---	---	---	---	---
	RESLC3	1.029-06	---	---	---	---	---	---	---	---
	REAC06	3.003-05	---	---	---	---	---	---	---	---
	REAC12	3.214-08	---	---	---	---	---	---	---	---
Operator Actuation of SSPS Signal	OS	---	3.638-07	3.638-07	---	---	---	---	---	---
Secure SI Per Operating Procedures Following SGTR	OP	---	---	---	---	---	9.605-05	---	---	---
Various Human Failures in Accident Recoveries	Total	2.178-06	2.326-08	2.326-08	7.686-06	---	---	---	1.827-08	1.582-05
	ZHESV3	4.440-07	5.985-09	5.985-09	---	---	---	---	---	---
	ZHEHS5	---	---	---	---	---	---	---	-	---
	ZHEAW4	---	---	---	---	---	---	---	---	---
	ZHERP2	---	---	---	6.677-06	---	---	---	---	1.582-05
	ZHESW1	2.457-07	---	---	---	---	---	---	---	---
	ZHERE2	2.217-07	---	---	---	---	---	---	--	---
	ZHEAW3	---	---	---	1.009-06	---	---	---	---	---
	ZHEF06	1.267-06	---	---	---	---	---	---	--	-
	ZHEOB2	---	1.727-08	1.727-08	---	---	---	---	1.827-08	--

Table 4
Unnormalized Fussel-Vesely Importances of Support System - Support System Pairs

Support Systems (Top Events)	Support Systems: Diesel Generator Systems	Instrument AC Power	Auxiliary Saltwater	Vital 125V DC, Unit 1	Component Cooling Water	Vital AC Power, Unit 1	Control Room Ventilation	Solid State Protection System	480V Switchgear Ventilation	Reactor Protection System	Vital AC & DC Unit 2
Non-Vital Electric Power (OG)	1.309-06	---	---	---	---	---	---	---	---	---	---
Diesel Generator Systems (GF,GG,GH,TG, TH,SW,FO)		3.462-07	1.548-06	2.687-06	3.981-06	1.661-06	2.231-06	3.691-07	1.590-06	---	7.511-07
Instrument AC Power (I1,I2,I3,I4)			---	---	---	---	---	---	---	---	---
Auxiliary Saltwater (AS)				3.050-07	---	2.994-07	3.507-07	---	---	---	---
Vital 125V DC, Unit 1 ((DF,DG,DH)					3.606-06	---	---	---	5.771-07	---	---
Component Cooling Water (CC)						9.814-07	3.728-07	2.55-07	---	--	- -
Vital AC Power, Unit 1 (AF, AG,AH)							---	---	5.676-07	---	---
Control Room Ventilation (CV)								---	---	---	---
Solid State Protection System (SA,SB)									---	1.032-06	-
480V Switchgear Ventilation (SV)										---	- -
Reactor Protection System (RT)											---

Table 3 (Continued) (Sheet 4 of 4)

System/Operator Action	Associated Top Events Or Their Total	LLOCA IF=2.02-04	SLBI IF=4.63-04	SLOCI IF=1.61-02	LOSWV IF=6.29-05	LOCV IF=7.99-02	LOPF IF=1.21-04	IMSIV, IF=1 07 01 TLMFW, IF=9 98 02 LCV, IF 8 73 02	SLOCN IF=5.26 03	ISI IF=7 39 02
	LB	5.925-04	---	6.491-05	---	---	---	---	5.539-05	---
	LV	---	---	---	---	---	---	---	---	---
	RW	---	---	---	---	---	---	---	3 940 05	- -
	VA	---	---	---	---	---	---	---	- -	--
	VB	---	---	---	---	---	---	---	--	--
	AC	6.270-03	---	---	---	---	---	---	-	---
	MU	---	---	---	---	---	---	---	9 236 05	-
ECCS, High Pressure	Total	4.930-03	---	---	3.060-03	---	---	---	3 697-05	---
	CH	---	---	---	---	---	---	---	---	---
	SI	---	---	---	3.060-03	---	---	---	---	---
	HR	---	---	---	---	---	---	---	---	---
*(RF4)	RF*	4.930-03	---	---	---	---	---	---	3.697-05	---
Reactor Vessel Integrity	VI	---	5.281-04	---	---	---	---	---	---	--
Turbine Trip & Main Steam Isolation	MS	---	1.040-03	---	---	---	---	---	---	---
Isolation of Ruptured SG	SL	---	---	---	---	---	---	---	---	---
Containment Isolation	CI	---	---	---	3.838-03	---	---	---	---	--
Containment Spray	Total	---	---	---	---	---	---	---	---	--
	CS	---	---	---	---	---	---	---	---	---
	SR	---	---	---	---	---	---	---	---	---
Operator/Recovery Actions										
Maintain Control for Hot Standby After an Accident	HS	---	---	---	---	5.010-06	5.010-06	5.010-06	---	5 010 06
Operator Trips RCPs After Loss of CCS to Prevent Seal LOCA	RP	---	2.700-03	---	---	---	---	---	---	---
Actions Needed to Maintain RCP Seal Cooling	SE	---	---	---	---	---	---	---	---	---
Electric Power Recovery Factors	Total	---	---	---	---	---	---	---	---	---
	RESLC1	---	---	---	---	---	---	---	---	---
	RESLC2	---	---	---	---	---	---	---	---	---
	RESLC3	---	---	---	---	---	---	---	---	-
	REAC06	---	---	---	---	---	---	---	---	---
	REAC12	---	---	---	---	---	---	---	---	---
Operator Actuation of SSPS Signal	OS	---	---	---	---	---	---	---	---	---
Secure SI Per Operating Procedures Following SGTR	OP	---	---	---	---	---	---	---	---	---
Various Human Failures in Accident Recoveries	Total	---	---	---	3.648-03	---	---	---	---	---
	ZHESV3	---	---	---	3.648-03	---	---	---	---	-
	ZHEHS5	---	---	---	---	---	---	---		- -
	ZHEAW4	---	---	---	---	---	---	---	---	--
	ZHERP2	---	---	---	---	---	---	---	-	
	ZHESW1	---	---	---	---	---	---	---	- -	
	ZHERE2	---	---	---	---	---	---	---		
	ZHEAW3	---	---	---	---	---	---	---		
	ZHEFO6	---	---	---	---	---	---	---		
	ZHEOB2	---	---	---	---	---	---	---	- -	

Table 3 (Continued) (Sheet 3 of 4)

System/Operator Action	Associated Top Events Or Their Total	Initiator, Initiator Frequency (yr⁻¹) LLOCA IF=2.02-04	SLBI IF=4.63-04	SLOCI IF=1.61-02	LOSWV IF=6.29-05	LOCV IF=7.99-02	LOPF IF=1.21-04	TMSIV, IF=1.07-01 TLMFW, IF=9.98-02 LCV, IF=8.73-02	SLOCN IF=5.26-03	IST IF=7.39-02
Support Systems										
Non-Vital Electric Power	OG	---	---	---	---	---	---	---	---	
Diesel Generator System	Total	---	---	---	---	---	---	---	---	
a. Unit 1 DGs	GF	---	---	---	---	---	---	---	---	
	GG	---	---	---	---	---	---	---		
	GH	---	---	---	---	---	---	---		
b. Unit 2 DGs	TG	---	---	---	---	---	---	---		
	TH	---	---	---	---	---	---	---		
c. Swing Diesel Align	SW	---	---	---	---	---	---	---	---	
d. Diesel Fuel Oil Transfer	FO	---	---	---	---	---	---	---	---	
Instrument AC Power	Total	---	---	---	---	---	---	---	2.346-05	
	I1	---	---	---	---	---	---	---	---	
	I2	---	---	---	---	---	---	---	---	---
	I3	---	---	---	---	---	---	---	2.346-05	
	I4	---	---	---	---	---	---	---	---	
Auxiliary Saltwater	AS	---	---	---	---	---	---	---	---	---
Vital 125V DC Power, Unit 1	Total	---	1.406-03	2.868-05	---	1.091-06	1.091-06	1.091-06	---	1.091-06
	DF	---	---	---	---	---	---	---	---	---
	DG	---	7.050-04	1.438-05	---	2.284-07	2.284-07	2.284-07	---	2.284-07
	DH	---	7.010-04	1.430-05	---	8.622-07	8.622-07	8.622-07	---	8.622-07
Component Cooling Water	CC	---	---	1.880-05	---	---	---	---	---	---
Vital AC Power, Unit 1	Total	---	6.892-04	2.824-05	---	---	---	---	---	---
	AF	---	---	---	---	---	---	---	---	---
	AG	---	---	1.412-05	---	---	---	---	---	---
	AH	---	6.892-04	1.412-05	---	---	---	---	---	---
Control Room Ventilation	CV	---	---	---	---	Initiator	---	---	---	---
Solid State Protection System	Total	9.622-04	1.829-03	---	1.506-02	---	---	---	---	---
	SA	9.622-04	1.443-03	---	7.580-03	---	---	---	---	---
	SB	9.622-04	1.426-03	---	7.480-03	---	---	---	---	---
480V Switchgear Ventilation	SV	---	---	---	Initiator	1.710-06	1.710-06	1.710-06	---	1.710-06
Reactor Protection System	RT	---	---	---	---	6.580-06	---	---	---	
Vital AC and DC Power, Unit 2	Total	---	---	---	---	---	---	---	---	
	BF	---	---	---	---	---	---	---	---	---
	BG	---	---	---	---	---	---	---	---	---
	BH	---	---	---	---	---	---	---	---	---
Frontline Systems										
Auxiliary Feedwater System	AW	---	1.222-03	---	---	2.169-06	2.169-06	2.169-06	---	2.169-06
Primary RCS Pressure Relief	Total	---	1.482-03	3.641-05	---	1.078-06	1.078-06	1.078-06	---	
	PR	---	---	3.641-05	---	---	---	---		
	OB	---	---	---	---	1.078-06	1.078-06	1.078-06		
ECCS, Low Pressure	Total	6.863-03	---	9.333-05	---	---	---	---	1.552-04	
	LA	5.925-04	---	6.483-05	---	---	---		7.805-05	

Table 5

Unnormalized Fussel-Vesely Importances of Frontline System - Support System Pairs

Frontline Systems (Top Events)	Support Systems (Top Events)												
	Non-Vital Electric Power (OG)	Diesel Generator Systems (GF, GG, GH, TG TH, SW, FO)	Instrument AC Power (I1, I2, I3, I4)	Auxiliary Saltwater (AS)	Vital 125V DC, Unit 1 (DF, DG, DH)	Component Cooling Water (CC)	Vital AC Power, Unit 1 (AF, AG, AH)	Control Room Ventilation (CV)	Solid State Protection System (SA, SB)	480V Switchgear Ventilation (SV)	Vital Reactor Protection System (RT)	AC & DC, Unit 2 (BF, BG, BH)
Auxiliary Feedwater System (AW)	3.960-07	1.487-05	9.690-06	---	9.450-06	9.120-07	4.596-06	6.347-07	1.439-07	4.530-07	---	4.218-07
Primary RCS Pressure Relief (PR, OB)	---	1.674-05	1.057-05	4.453-07	1.617-06	1.782-06	1.714-06	8.[illegible]19-07	3.884-07	---	---	---
ECCS, Low Pressure (LA, LB, LV, RW, VA, VB, AC, MU)	---	4.632-06	1.234-07	---	4.618-07	---	4.929-07	2.305-07	---	---	---	---
ECCS, High Pressure (CH, SI, HR, RF*)	---	3.647-06	---	---	---	---	3.243-08	---	---	2.465-07	---	---
Reactor Vessel Integrity (VI)	---	1.241-06	2.798-07	---	---	---	---	---	---	---	---	---
Turbine Trip & Main Steam Isolation (MS)	---	1.612-07	8.315-07	3.050-07	2.214-06	5.180-07	3.424-07	---	1.551-06	---	---	---
Isolation of Ruptured SG (SL)	---	---	3.849-07	---	2.354-07	---	---	---	---	---	---	---
Containment Isolation (CI)	---	9.309-08	---	---	---	---	---	---	---	---	---	---
Containment Spray (CS, SR)	---	---	---	---	---	---	---	---	---	---	---	---
Interfacing LOCA Event Tree Top Events (VO, VC, VR, SM, IT, LW, ME)	---	---	---	---	---	---	---	---	---	---	---	---

*RF does not include RF4 as RF4 is a post-core melt action.