JAN 30 1985

MEMORANDUM FOR: Darrel G. Eisenhut, Director
Division of Licensing

FROM: Themis P. Speis, Director
Division of Safety Technology

SUBJECT: INITIAL REVIEW OF THE SEABROOK STATION PROBABILISTIC SAFETY ASSESSMENT

We have completed our initial review of the Seabrook Probabilistic Safety Assessment (PSA). The Seabrook PSA, performed for Public Service of New Hamphsire by Pickard Lowe and Garrick, Inc. (PL&G) was voluntarily submitted to NRC for information. The PSA considers both internal and external initiating events, containment failure modes, radioactive releases, and associated consequences. The Reliability and Risk Assessment Branch contracted technical assistance from Lawrence Livermore National Laboratories (LLNL) in reviewing the internal and external events analyses leading to plant damage states (core damage or core melt). The Reactor Systems Branch (RSB) in the Division of Systems Integration is providing a review of the containment failure modes and consequence analysis, with the assistance of Brookhaven National Laboratory (BNL), and the results of their preliminary review will be provided separately.

We and our contractors believe that the Seabrook PSA is an extremely comprehensive study with advances in the state-of-the-art in terms of identifying initiating events and their groupings, the inclusion of support states in event trees, and quantification of accident sequence frequencies, plant damage state frequencies and resultant overall frequency of core melt. However, some of these aspects make the PSA less amenable to estimating the effect on the results from changes in models and obtaining qualitative engineering insights into the strengths and weaknesses of plant design and operation.

The Seabrook PSA estimate of overall core melt frequency is about $2x10^{-4}$/reactor-year. However, a very large number of sequences contribute to the overall core melt probability with the single most dominant sequence contributing less than 15% to the total and the top 22 sequences contribute

FOIA-87-CC6

B/38

Contact: Sarah Davis, RRAB
49-27546

8502070262 XA (23pp)

OFFICE ▶							
SURNAME ▶							
DATE ▶							

approximately 50% to the total. The Seabrook PSA included consideration and quantification of 58 initiating events. These were collapsed to initiating event groups and the contribution to core melt frequency by these initiator groups are displayed in Tables 1 through 3 in Enclosure 1, Summary Review of the Seabrook PSA, as well as delineation of the top 22 accident sequences. Table 4 in Enclosure 1 displays the rough aggregation of accident sequences with similar characteristics to better identify dominant contributors to core melt frequency. As can be seen, sequences initiated by Loss of Offsite Power overwhelmingly dominate core melt frequency followed by Fire initiated and Small LOCA sequences with significantly smaller contributions. These groups of sequences, along with Seismic Events, Component Cooling Water System Failures and ATWS sequences are discussed in Enclosure 1.

Our review of the Seabrook PSA did not identify any safety issues which merit immediate action. We have enclosed a summary of the salient features of the review, with regard to core melt, of the Seabrook PSA (Enclosure 1) and Enclosure 2, the draft report from LLNL. Overall, the review did not identify a discrepancy or error which is estimated, at this point, to significantly change the quantitative results of the PSA. The areas of disagreement and questions are documented in the review report (Enclosure 2).

The review has been impeded by circumstances and problems in several areas. This PSA was submitted to the NRC voluntarily by Public Service of New Hampshire during a period of severe financial problems surrounding the completion of the Seabrook plants and the stability of the applicant utility. Since the PSA has not been tied to a specific licensing action, the applicant made the decision that they were not able to allot resources for the support of the review of this document. They did provide personnel to conduct a plant visit in late August, but did not provide any further support in terms of supplying documentation requested, answers to questions from the Lab and the staff, and, having severed their contract with their consultant who performed the PSA, could not provide an avenue for answers or documentation from the authors of the PSA. We acknowledge that these decisions were not made in a spirit of non-cooperation but rather financial circumstances which outweighed a decision to provide support. However, this support is important in conducting a PSA review in order to arrive at meaningful conclusions regarding the validity of the results and the insights gained during the performance of the PSA.

OFFICE							
SURNAME							
DATE							

With this evaluation, the Phase I (preliminary review) work, as identified in the FY84 Operating Plan (Item V.B.1.a.iv), on the Seabrook PSA is complete.

Original Signed By

Themis P. Speis, Director
Division of Safety Technology

Enclosures:
1. Staff Summary Review of the Seabrook PSA
2. A Review of the Seabrook Station PSA

cc: H. Denton
E. Case
R. Vollmer
R. Bernero
H. Thompson
G. Knighton w/o Enclosure 2
V. Nerses " " "
B. Sheron " " "
W. Lyon " " "
D. Silver " " "
S. Brocoum " " "
P. Sobel " " "

Distribution

Central file
RRAB Rdg
SDavis
Davis CHRON
RFrahm
AThadani
FRowsome
AD/TRg
TSpeis
DST CHRON

OFFICE	RRAB:DST	RRAB:DST	RRAB:DST	AD/T:DST	D:DST		
SURNAME	SDavis mT	RFrahm	AThadani	FRowsome	TSpeis		
DATE	1/25/85	1/25/85	1/29/85	1/29/85	1/29/85		

NRC FORM 318 (10 80) NRCM 0240 OFFICIAL RECORD COPY USGPO 1981-

ENCLOSURE 1

Staff Summary Review of the Seabrook PSA

BACKGROUND

Lawrence Livermore National Laboratory (LLNL) conducted a review of the Seabrook Station Probabilistic Safety Assessment for the Reliability and Risk Assessment Branch (RRAB), Division of Safety Technology. This probabilistic safety assessment (PSA) was performed by Pickard, Lowe, and Garrick Inc. for Public Service of New Hampshire (PSNH) and Yankee Atomic Electric Company (YAEC). The PSA was also provided to the NRC for its information. The review of the Seabrook PSA was performed by a project team composed of personnel from NRC staff, LLNL staff, subcontractors and consultants.

OVERALL EVALUATION OF THE SEABROOK PSA

An examination and review of the dominant sequences identified in the PSA was performed in light of the various concerns that have been identified in Chapters 3 and 4 of Enclosure 2 for the internal and external events. This examination was necessarily limited by the inability to reconstruct and reevaluate the event trees with consideration of these concerns, and to then compare the new results to the PSA results. It was not possible to perform this evaluation and comparison because of the lack of information and answers to technical questions.

Nevertheless, the overall findings of the LLNL review resulted in the judgment that the dominant sequences presented in the PSA generally appear to be reasonable (although conservative) in a quantitative sense. That is to say that one could expect to find that the quantitative results of a new evaluation would not find the probability of core melt to be significantly larger than described in the PSA, because of the generally conservative quantitative approaches and assumptions incorporated in many places in the PSA. A summary of the sequences and review findings is provided in the following section. No significant omissions were found in terms of an overall contribution to the frequency of core melt. Several modeling errors were found that indicate an incomplete or different understanding of interactions between plant sytems or human beings (operators) and plant systems; these are described in the internal events section of Enclosure 2. While LLNL considered it likely that a reevaluation of the sequences where these differences were identified would not significantly affect the overall core melt frequency, it may provide different qualitative results regarding the understanding of dominant contributors and therefore different insights. However, the significance of these differences could not be completely assessed.

SUMMARY

The PSA results for core melt probabilities were 1.6E-4 per reactor year (RY) for internal events and 6.2E-5/RY for external events, for a total of 2.3E-4/RY. External events were dominated by contributions of 2.9E-5/RY from seismic events and 2.6E-5/RY for fires. The scope of the review did not include a review of containment response or offsite consequences nor extensive requantification.

A very large number of sequences contribute to the overall core melt frequency

The Seabrook PSA included consideration and quantification of 58 events. Tables 1, 2, and 3 display the accident sequence contributions to core melt frequency grouped by initiating event categories and provide a summary of the top 22 sequences.

However, by collapsing these sequences to categories of sequences with similar characteristics, the resultant sequence frequencies and contributions to overall core melt frequency are presented in Table 4.

TABLE 1 CONTRIBUTIONS OF ACCIDENT SEQUENCES
GROUPED BY INITIATING EVENT TO CORE MELT FREQUENCY

Initiator Group	Percent Contribution	Contributors	Percent Contribution
Loss of Coolant Inventory	11	• Small LOCA • Others	8 3
Transient Events (excluding support system faults)	31	• Reactor trip • Turbine trip • Partial loss of feedwater • Steamline break • Excessive feedwater • MISV closure • Others	6 4 4 4 3 2 8
Common Cause Initiating Events	58	• Loss of offsite power • Seismic events • Fires • Other support system faults • Other external events	29 12 11 3 3
Total	100		100

Core Melt Frequency: Mean 2.3×10^{-4} per reactor year

TABLE 2 CONTRIBUTIONS OF SEQUENCES GROUPED BY INITIATING EVENT TO FREQUENCY OF RISK SIGNIFICANT RELEASE CATEGORIES AND CORE MELT

Initiating Event Group	Initiating Event	Release Category Frequency Contributions (events per reactor year)						Core Melt Frequency Contributions (events per reactor year)
		Large Containment Bypass $\overline{S6V}$	Small Containment Bypass $\overline{S2V}$	Basemat Melt-Through $\overline{S4V}$	Late Overpressure with Vaporization Release $\overline{S3V}$	Late Overpressure - No Vaporization Release S3	Containment Intact S5	
Loss of Coolant Inventory	Large LOCA	•	•	•	•	•	1.1-6	1.4-6
	Small LOCA	•	•	•	•	2.5-6	1.7-5	2.0-5
	Interfacing Systems LOCA	1.8-6	0	0	0	0	0	1.8-6
	Steam Generator Tube Rupture	•	•	1.0-7	8.2-7	•	6.5-7	1.7-6
General Transients	Reactor Trip	•	•	•	•	6.3-6	6.2-6	1.3-5
	Turbine Trip	•	•	•	•	3.9-6	3.9-6	1.0-5
	Loss of Main Feedwater	•	•	1.0-6	8.3-6	•	•	1.1-5
	Partial Feedwater Loss	•	•	•	•	5.0-6	2.5-6	7.8-6
	Excessive Feedwater	•	•	•	•	2.8-6	2.7-6	5.7-6
	Loss of Condenser Vacuum	•	•	•	•	•	8.9-7	1.1-6
	MSIV Closure	•	•	•	•	•	4.9-6	5.0-6
	Loss of Primary Flow	•	•	•	•	1.2-6	1.1-6	2.4-6
	Steam Line Break	•	•	•	•	•	6.9-6	7.3-6
	Main Steam Relief Opens	•	•	•	•	5.0-7	1.4-7	7.8-7
Common Cause Initiating Events - Support System Faults	Loss of Offsite Power	•	•	6.8-6	5.5-5	4.9-6	1.5-6	6.9-5
	Loss of One DC Bus	•	•	•	•	•	1.7-6	2.3-6
	Loss of Service Water	•	•	•	•	2.5-6	0	2.5-6
	Loss of Component Cooling	•	•	•	•	1.4-6	0	1.4-6
- External Events	Seismic Events (total)	5.8-7	1.7-5	5.3-7	4.0-6	2.9-6	2.8-6	2.8-5
	Fires (total)	•	•	6.6-7	5.3-6	2.0-5	•	2.5-5
	Flood (total)	•	•	2.3-7	1.9-6	1.8-6	•	3.9-6
	Truck Crash	•	•	1.8-7	1.4-6	1.4-7	•	1.8-6
Total		2.4-6	1.8-5	1.0-5	8.0-5	5.8-5	6.0-5	2.3-4

*Less than 1% contribution to release category frequency.

NOTE: Exponential notation is indicated in abbreviated form; i.e., 1.1-6 = 1.1×10^{-6}.

TABLE 3 SUMMARY OF ACCIDENT SEQUENCES WITH SIGNIFICANT RISK AND CORE MELT FREQUENCY CONTRIBUTIONS

Initiating Event	Additional System Failures/ Human Actions	Resulting Dependent Failures	Sequence Frequency (per reactor year)	Sequence Ranking		
				Core Melt	Latent Health Risk	Early Health Risk
Loss of Offsite Power	Onsite AC Power, No Recovery of AC Power Before Core Damage	Component cooling, high pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	3.3-5	1	1	*
Loss of Offsite Power	Service Water, No Recovery of Offsite Power	Onsite AC power, component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	9.2-6	2	2	*
Small LOCA	Residual Heat Removal	None.	8.9-6	3	*	*
Control Room Fire	None	Component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	8.7-6	4	3	*
Loss of Main Feedwater	Solid State Protection System	Reactor trip, emergency feedwater, high and low pressure makeup (ECCS), containment filtration and heat removal.	8.3-6	5	4	*
Steam Line Break Inside Containment Heat Removal	Operator Failure to Establish Long Term		5.6-6	6	*	*
Reactor trip	Component Cooling	High and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	4.6-6	7	5	*
Loss of Offsite Power	Train A Onsite Power, Train B Service Water, No Recovery of AC Power Before Core Damage	Train B onsite power, component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	4.4-6	8	6	*
Loss of Offsite Power	Train B Onsite Power, Train A Service Water, No Recovery of AC Power Before Core Damage	Train A onsite power, component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration and heat removal.	4.4-6	9	7	*
PCC Area Fire	None	Component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration, and heat removal.	4.1-6	10	8	*

*Negligible contribution to risk.

NOTE: Exponential notation is indicated in abbreviated form; i.e., 3.3-5 = 3.3×10^{-5}.

0989P122283

TABLE 3 (continued)

Initiating Event	Additional System Failures/ Human Actions	Resulting Dependent Failures	Sequence Frequency (per reactor year)	Sequence Ranking		
				Core Melt	Latent Health Risk	Early Health Risk
Partial Loss of Main Feedwater	Component Cooling	High and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration, and heat removal.	3.8-6	11	9	*
Cable Spreading Room Fire	None	Component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration, and heat removal.	3.5-6	12	10	*
Loss of One DC Bus	Emergency Feedwater, No Recovery of Emergency or Startup Feedwater	Bleed and feed cooling, Train A containment filtration and heat removal.	3.2-6	13	*	*
Reactor Trip	Operator Failure to Establish Long Term Heat Removal.	None.	3.0-6	14	*	*
Turbine Trip	Component Cooling	High and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration, and heat removal.	2.8-6	15	11	*
Loss of Service Water	None	Component cooling, high and low pressure makeup, reactor coolant pump seal LOCA, containment filtration, and heat removal.	2.3-6	16	12	*
Partial Loss of Feedwater	Operator Failure to Establish Long Term Heat Removal	None.	2.3-6	17	*	*
Turbine Building Fire	Onsite AC Power, No Recovery of AC Power Before Core Damage	Offsite power, component cooling, high and low pressure makeup (ECCS), reactor coolant pump seal LOCA, containment filtration, and heat removal.	2.3-6	18	13	*
Small LOCA	Train B Safety Features Actuation, Train A Residual Heat Removal	Train A high and low pressure makeup and residual heat removal; train B containment filtration and heat removal.	2.2-6	19	*	*
Small LOCA	Train A Safety Features Actuation, Train B Residual Heat Removal	Train B high and low pressure makeup and residual heat removal; train B containment filtration and heat removal.	2.2-6	20	*	
Turbine Trip	Reactor Trip, Failure to Manually Scram Reactor and to Effect Emergency Boration	Functional inability to provide adequate high pressure makeup.	1.9-6	25	*	*
Interfacing Systems LOCA	None	Low pressure makeup, residual heat removal, containment isolation and filtration.	1.8-6	27	14	1

*Negligible contribution to risk.

NOTE: Exponential notation is indicated in abbreviated form; i.e., 3.8-6 = 3.8×10^{-6}.

TABLE 4

Initiating Event Group	Aggregate Frequency/RY	Contribution to Overall CMF	Number of Sequences in Top 22	Total Frequency/RY in Top 22
Small LOCA	$\sim 2\times10^{-5}$	9%	3	$\sim 1\times10^{-5}$
Loss of Offsite Power	$\sim 7\times10^{-5}$	30%	4	$\sim 5\times10^{-5}$
Seismic	$\sim 3\times10^{-5}$	13%	0	
Fire	$\sim 3\times10^{-5}$	13%	4	$\sim 2\times10^{-5}$
Component Cooling Water System Failure	$\sim 1\times10^{-5}$	4%	3	$\sim 1\times10^{-5}$
ATWS	$\sim 1\times10^{-5}$	4%	2	$\sim 1\times10^{-5}$
Transient (other than LOOP) followed by Failure of Long Term Heat Removal	$\sim 1\times10^{-5}$	4%	3	$\sim 1\times10^{-5}$
Interfacing Systems LOCA	$\sim 2\times10^{-6}$	~1%	1	$\sim 2\times10^{-6}$
Other Transients	$\sim 4\times10^{-5}$	19%	2	$\sim 5\times10^{-6}$
Other External				
Flood, Truck Crash	$\sim 6\times10^{-6}$	~3%	0	

SEQUENCES INITIATED BY INTERNAL EVENTS

The extent and type of internal event initiators and their treatment is generally reasonable and consistent with those considered in other PRAs.

The event tree models in most cases correctly represented the accident sequence phenomenology assumed in the PSA; however, several areas of disagreement with the assumed phenomenology were identified. There is also a concern that the requirement to have each event on an event tree independent of the others has resulted in large and very complex trees which are difficult to follow and analyze. In addition, the large number of sequences, on the order of 100 times as many as in previous PSAs, effectively fragmented many accident scenarios which could be simply described as single sequences into a large number of sequences, so that the usefulness of the event tree sequences as a means to obtain engineering insights was lost. Although many deficiencies in these trees are described in the text of Enclosure 2, it was not possible to provide a preliminary assessment of the quantitative effect on the PSA results, primarily because of the complexity of the trees in the PSA and their use of proprietary codes to perform the quantitative evaluations.

LOSS OF OFFSITE POWER

Sequences initiated by a loss of offsite power, taken collectively, dominate the overall core melt frequency. Of the sequences comprising 30% of core melt frequency, 4 sequences are in the top 22 sequences with a 22%

contribution to CMF, the highest of which, contributes 15%. In this sequence, loss of offsite power is followed by failure of onsite AC power (2 of 2 diesel generators failing) with a resultant Reactor Coolant Pump Seal failure (RCP Seal LOCA) due to loss of Component Cooling Water and High Pressure Makeup. The other three sequences initiated by loss of offsite power involve total Service Water System failure and combinations of one train of Service Water failure with failure of the opposite train of onsite AC power. These sequences also lead to a RCP seal LOCA, though the individual sequences have lower frequencies due to the differing system failures in the accident progression. The frequency assumed for loss of offsite power is generally consistent with those in other PSAs, but two assumptions in these sequences cause concern. The first is that the RCP seal LOCA will occur immediately upon loss of all AC power and that the leak rate is 20 gpm per pump. Based on previous analyses (Millstone 3 PRA, for example) it is not reasonable to assume that a leak will occur immediately. A more realistic time frame is on the order of 30 to 60 minutes before the seals fail. At that time, previous analyses assume that the leak rate will rapidly accelerate to a much higher rate (~300 gpm) and subsequent total failure since the seals will be in a degraded condition under high mechanical and thermal stress. The extremely low flow rate assumed in the Seabrook PSA extends the occurrence of core uncovery and damage to a much later time than is considered realistic. Under the assumption of a higher flow rate, core damage is more realistically expected to occur about two hours after power is lost and unrestored. Justification for the assumptions in the PSA was not provided and the overall effect of the concerns regarding them have not yet been quantitatively assessed. LLNL considered it likely that a reevaluation of these sequences where differences were identified would not significantly affect overall core melt frequency, however, it may result in a different understanding of the timing of core melt and the resultant distribution of sequences over plant damage states.

SMALL LOCA

Of the sequences initiated by a small LOCA comprising 9% of overall core melt frequency, 3 sequences appear in the top 22 with a contribution of about 4% to CMF. The highest frequency sequence involves failure of Residual Heat Removal (RHR) as a source of low pressure injection following manual depressurization with the other two involving combinations of a Safety Features Actuation (SFA) Signal train failure and failure of an RHR train. The SFA failure affects containment filtration and heat removal. Of concern are the two separate values assumed for a small LOCA, representing breaks that can be isolated and those which are nonisolable. It has been generally recognized in previous PSA analyses that isolable breaks do not significantly contribute to overall small LOCA frequency due to the amount of time available for the operator to isolate them prior to the need for emegency core cooling. Therefore, the concern is with the nonisolable break frequency assumed in the Seabrook PSA which is lower than those found in various other PSAs and PSA reviews. This frequency is based on the ability to isolate a random RCP seal LOCA at a given plant. The Seabrook plant does not have primary loop isolation valves, thus an RCP seal LOCA would be considered nonisolable. Since this was not considered in the PSA frequency on nonisolable small LOCAs, it appears that this initiating event has been underestimated by as much as a factor of four when compared to other data sources (e.g., ANO-1 IREP analyses).

Component Cooling Water System Failure (CCW)

The other category of sequences of interest involve failure of the CCW System. These three sequences, which involve resultant RCP seal failure, have an aggregate frequency of $\sim 1\times10^{-5}$/RY and contribute 5% to overall CMF. This value of 1×10^{-5}/RY is somewhat lower than those determined for other PRAs for similar plants (e.g., Zion, Indian Point). It has not yet been determined whether the particular configuration of the CCW system at Seabrook has design features which would explain this difference, one aspect of the PSA worth noting is that while the study considers a total loss of the CCW System as an initiating event, it does not consider loss of a single train. The basis provided is that if a single train is lost, the reactor will not immediately trip and the operator can proceed with an orderly shutdown, thus it is not an initiating event.

Anticipated Transients Without Scram

ATWS sequences contribute $\sim$4% to overall core melt frequency. Two sequences, with collective frequency of $\sim 1\times10^{-5}$/RY, were among the top 22 sequences. One sequence is initiated by a loss of main feedwater with subsequent mechanical failure of the Solid State Protection System (8.3×10^{-6}/RY) and the other is initiated by a Turbine Trip followed by failure of automatic and manual scramming of the reactor (1.9×10^{-6}/RY).

Using the new ATWS rule to provide guidance and information, some problems with the ATWS event tree were identified, in areas such as operator recovery and credit for bleed and feed.

The PSA gives credit to the possibility of operator action to effect manual reactor scram following automatic scram failure. This action, however, is not modeled explicitly on the tree; it is applied directly to the failure of RPS leading to ATWS. It is valid to consider this type of recovery, but an action of this import should have been included explicitly on the tree. It is also important to note that this recovery action can only be applied to electrical failures of the RPS, so that RPS failures should have been divided into electrical and mechanical failures as stated in the ATWS rule.

The assumption that it is necessary for the operator to shut down the reactor after the initial phase of the ATWS is reasonable and consistent with the ATWS Rule. However, the Seabrook PSA assumes that this action must be taken within ten minutes, which appears to be conservative. Once the initial phase of the ATWS is over, the power equilibrates at the secondary heat demand and the plant will operate safely for extended periods of time. This was supported by many analyses and a simulator run performed on the Seabrook simulator during the plant visit of August 29-31, 1984. It appears that the time frame is more on the order of 60 minutes or more, except when a primary safety valve sticks open or the ATWS tree is entered from a LOCA initiator. In this case, a 20 minute time frame is more appropriate than the 10 minute operator action time assumed in the Seabrook PSA.

The PSA also assumed that it is possible to mitigate an ATWS by using bleed-and-feed with HPI alone if emergency feedwater fails. This would theoretically provide boration to shut down the reaction simultaneously with bleed-and-feed cooling. This method has not been considered in most other PSAs, and takes an inordinately large amount of credit for the ability of HPI to provide flow at operating pressure. Also, there would be much greater amounts of heat to be removed through the PORVs with makeup flow than for a normal bleed-and-feed scenario. It is not clear that heat removal and reactor shut down could be accomplished under these conditions without the emergency feedwater system. Therefore, it is more appropriate to conclude that all sequences with failure of emergency feedwater would lead to core melt and may increase the ATWS sequence frequency. This may be offset by the other conservatisms assumed in evaluating ATWS events. The overall quantitative effect on ATWS and overall core melt frequency has not yet been evaluated.

Transients with Failure of Long Term Heat Removal

The contribution to overall core melt frequency from this category of sequences comes from a large number of sequences, only three of which appear in the top 22 sequences. Failure of long term heat removal (high and low pressure makeup in the RHR mode configuration) is dominated by common cause failures (e.g. maintenance unavailabjlities) and independent hardware failures. These failures involve failure of pumps to start and valves in RHR trains and heat exhanger valves failing to open.

The category of "other transients" is comprised of an extremely large number of sequences of the remaining sequences analyzed as contributors to overall core melt frequency. This category has not been decomposed to obtain insights in this initial review.

The functional success criteria used in the PSA were generally found to be reasonable, with some exceptions. These criteria, however not clearly stated in many cases, included both conservative and optimistic examples and in general appeared to be inadequately documented.

The review of the failure rate data used in the PSA consisted of a comparison of the individual component failure rates with other sources and a review of system failure probabilities and unavailabilities. The data values presented were found to be reasonably consistent with other data sources available to the review. A comparison of system failure probabilities with other sources of similar data revealed that these values were reasonably consistent with the other sources.

Consideration and treatment of dependencies and common cause failures in the PSA were evaluated in the review in three categories: common cause initiating events, intersystem dependencies, and intercomponent dependencies. The methodology used in the analysis appears reasonable and appropriate. No important omissions in the treatment of dependencies were identified by the review. The treatment of common cause data was of some concern because of the exclusion of passive components and the use of very low beta factors (i.e., factors to account for common cause failures) for some components although no instance was identified that would significantly change the results.

EXTERNAL EVENTS

The external events considered in the PSA are earthquakes, fires, aircraft accidents, internal and external flooding, extreme winds, and turbine missiles. It is important to note that sequences initiated by the various external events (not including LOOP) were not significant contributors and that only fire initiated sequences appeared in the top 22 sequences. This is not entirely consistent with other PSA findings (such as those for Zion, Indian Point, and Millstone 3).

The methodologies used in the detailed assessments are generally reasonable and consistent with the state-of-the-art; however, there were notable disagreements in several areas.

EARTHQUAKES

SEISMIC HAZARD AND SEISMIC FRAGILITY.

Though sequences initiated by seismic events contribute ~13% to overall core melt frequency, none of these sequences appear in the top 22 and the highest individual sequence frequency is $\sim 4\times10^{-6}$/RY. The occurrence of a seismic event could initiate a sequence at the Seabrook plant in any of several ways. Failure of the offsite power transformers (Reserve Auxiliary Transformers and Unit Auxiliary Transformers), or switchyard equipment, would result in offsite power to the plant being lost. Also, at higher accelerations, a failure of the reactor internals could cause the control rods to jam and not position for reactor shutdown. It is also possible that an earthquake could cause the anchor bolts tying down the steam generators or reactor coolant pumps to fail, thereby permitting the equipment to tilt, possibly resulting in a break at the primary cooling system piping. Other failures such as instrument buses, that would cause a transient type event, would occur at accelerations higher than those that would already have caused a loss of offsite power and would result in the same sequences.

The methodology used in the evaluation of the frequency of the seismic hazard at Seabrook is consistent with the state-of-the-art of commercial PSAs. However, there is disagreement with numerous applications of the methodology in the PSA. In particular, the procedure used to perform the uncertainty analysis failed to document the choices made and the uncertainty assigned to key parameters in the analysis. A review of individual parameters in the analysis and a comparison with the interim Seismic Hazard Characterization Program lead to the qualitative conclusion that the hazard analysis results may be optimistic and the uncertainty underestimated.

The methodology used in the PSA for determining the seismic fragilities is appropriate and adequate to obtain a rational measure of the capacity of the structures and equipment. Based on a preliminary review of the results of the PSA, the mean frequency of core melt value of 2.89E-5 per year appears to be high relative to the optimistic hazard curves used in the analysis. Calculations indicate that the capacities of the key components at the SSE value are low and generally less than values determined for other PSAs such as Limerick and Millstone. In addition, comparing fragility parameter values of Seabrook and other PWRs (new and old), the capacity values of equipment considered also appeared to be low for Seabrook. Based on experience with past PSA reviews and information gained during the site inspection, the capacities of the dominant contributors, though they have not been quantitatively reevaluated, are likely not to be as low as indicated.

FIRES

Fire induced sequences contribute ~13% to overall core melt frequency with 4 sequences appearing in the top 22 sequences contributing $\sim 2 \times 10^{-5}$/RY (9%) to CMF. These sequence initiators are Control Room fire (8.7×15^{-6}/RY), Primary Component Cooling Area Fire (4.1×10^{-6}/RY), Cable Spreading Room Fire

($3.5x10^{-6}$/RY), and Turbine Building Fire followed by loss of all AC power ($2.3x10^{-6}$/Ry). All of these sequences result in loss of component cooling, RCP seal LOCA, loss of ECCS, and failure of containment filtration and heat removal. The fire analysis performed for the PSA, based on current fire protection guidance, appears accurate and valid and the frequencies of the fire induced initiating event which include system failure appear reasonable. The contribution to core damage due to fires at the various locations analyzed falls within the range of those calculated from other fire assessments at nuclear power plants (1E-4/RY to 1E-7/RY).

The analysis of fire propagation for determining the loss of safety related functions is rigorous and explicit and the considerations of fire phenomena, material properties, fire detection and suppression, operator action, and modeling uncertainty at each fire location were reasonable.

There is a concern, however, about the manner in which the fire induced initiating events are processed through the plant matrix. It appears that these initiating events, which already include component or system failures, are being incorrectly combined with auxiliary and front-line event trees that have not explicitly considered these same failures. This concern has yet to be verified and evaluated.

INTERNAL FLOODING

The PSA treats internal flooding primarily qualitatively, with a quantitative analysis performed for a turbine building and switchgear room flood. The qualitative analyses consider all internal flood sources for each defined location, including floods caused by fire protection equipment and sources from adjacent locations. In all these locations it was concluded that the risk due to flooding was insignificant. The quantification of the turbine building flood appears to be reasonable and adequate as was the qualitative treatment of flooding in other locations.