TECHNICAL REPORT
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TECHNICAL EVALUATION REPORT OF THE IPE SUBMITTAL AND RAI RESPONSES FOR THE WATERFORD-3 STEAM ELECTRIC STATION

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EXECUTIVE SUMMARY

This Technical Evaluation Report (TER) documents the findings from a review of the Individual Plant Examination (IPE) for the Waterford-3 Steam Electric Station (W3). The primary purpose of the review is to ascertain whether or not, and to what extent, the IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 and achieves the four IPE sub-objectives. The review utilized both the information provided in the IPE submittal and additional information provided by the licensee, Entergy Operations, Inc., in response (RAI Responses) to NRC requests for additional information (RAI).

E.1 Plant Characterization

The Waterford 3 Steam Electric Station is a 1153 MWe, 3410 MWth Combustion Engineering pressurized water reactor (PWR). The reactor coolant system (RCS) consists of the reactor vessel, two U-tube steam generators, 4 shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnected piping. The plant is operated by Entergy Operations, Inc., and started commercial operation in the Fall of 1985. There are no other operating units on site.

Design features at Waterford 3 that impact the core damage frequency (CDF) are as follows:

- There is no feed and bleed capability at this plant. No pressurizer PORV exists and the HPSI/charging pumps do not have the requisite head to lift the safety valves.

- The turbine driven main feedwater pumps will continue to run for most transients, as the pump flow output is automatically matched to the decay heat level.

- There are two motor driven (capacity 350 gpm each) and one turbine driven (capacity 700 gpm) EFW pump. In addition, a manually started AFW pump is also available, should the other three pumps fail (the AFW pump is normally used during startup/shutdown operations).

- The EFW control valves fail open on loss of instrument air, and there is also a backup nitrogen accumulator supply in case of loss of instrument air. The turbine driven EFW pump does not require room cooling (according to calculations, RAI responses), whereas the motor driven EFW pumps do.

- The DC battery (battery AB) supplying control to the TDEFW pump has a SBO depletion time of 4 hours with proceduralized load shedding (1 hour without load shedding), according to the submittal.

- Condensate pumps may be used to provide feedwater to the steam generators, provided the secondary system has been depressurized to 500 psia. There are three parallel condensate pumps. The condenser hotwells have enough inventory to supply the condensate pumps for 24 hours.

- There are two EDGs. The EDGs need cooling by CCW, ventilation by dedicated fans and DC power provided by the station batteries. A diesel compressor has been added to the plant post-IPE, to help in case of problems with startup compressed air.

- There is no service water system at this plant. Instead, the ultimate heat sink is provided by the dry cooling towers. As there are multiple fans in the towers, they can be maintained piecemeal, such that maintenance would not disable the whole tower (although in the IPE it is conservatively assumed that it does). Also, in case of increased demand (depending on air temperature) and during normal operation there are additional wet cooling towers which are used to increase the heat rejection capacity. The IPE assumes that the wet cooling towers are needed in case of a LOCA, when several types of safety equipment may be operating simultaneously.

- The CCW is needed to cool the HPSI pumps, the LPSI pumps, containment spray pumps, shutdown heat exchangers (also used for containment spray recirculation cooling), containment fans, the emergency diesel generators and the central chillers used to provide HVAC cooling for several plant areas.

- The instrument air system is necessary for operation of the MFW system and the normal pressurizer spray (but not the auxiliary spray, supplied by the charging pump). All the other important systems (EFW, CCW, ACCW, containment sump recirculation valves) are provided with a backup air or nitrogen accumulator system. There are two instrument air compressors, of which one is sufficient to supply the requisite loads in an intermittent type of operation. In case of failure of both compressors, a cross tie to the station air system automatically opens; the station air has three compressors. Therefore the compressed air system seems to be relatively reliable and the systems affected are relatively few.

- Room cooling or ventilation is needed for several important systems: HPSI [not needed during the refueling water storage pool (RWSP) injection phase due to the low temperature of the water pumped], LPSI (not needed during the injection phase), containment sprays (not needed in the injection phase), MDEFW pumps, normal pressurizer sprays, emergency diesel generators and the CCW pumps.

- The switchover to recirculation is automatic. However, the operator must manually close the RWSP suction valves at that time.

- The recirculation spray (using the CSS pumps aligned to the containment sump and the shutdown heat exchangers) is necessary to provide cooling of the containment sump water.

Other design features are discussed in Section 1.2.

The Waterford 3 Steam Electric Station utilizes a large dry containment. It is a freestanding steel vessel surrounded by a reinforced concrete shield building. Both the thermal power level and the containment free volume of Waterford 3 are similar to those of Zion.

The following plant-specific features are important for accident progression in the Waterford 3 plant:

- A cavity design which facilitates flooding of the reactor cavity. According to the IPE, water can readily flow from the containment sump to the reactor cavity. Flooding of the cavity is accomplished through a small tunnel that connects to the ductwork that provides reactor cavity cooling. Flooding of the reactor cavity and the low placement of the reactor vessel in the reactor cavity ensures that ex-vessel cooling can occur.

- A steel shell containment that is vulnerable to direct attack by dispersed core debris. However, based on the consideration of potential debris dispersing paths and MAAP calculations, the Waterford IPE discounts the possibility of direct corium attack on the steel containment wall.

- A reactor vessel with no lower head penetrations. This delays the time of vessel failure, but may cause a more energetic failure with larger hole size.

- The large amount of Zircalloy in the core assemblies. The amount of Zircalloy in the core assemblies of Waterford 3 is about 40% more than that of Zion. The amount of hydrogen produced during a severe accident is thus more for Waterford 3 than for Zion.

- A small reactor cavity with very little area for ejected core material to disperse to the upper containment region. The cavity is open to the upper compartment through a very small annulus between the vessel and cavity wall.

- The large containment volume, high containment pressure capability, and the open nature of compartments which facilitates good atmospheric mixing.

E.2 Licensee's IPE Process

The IPE was initiated in late 1988. The model reflects the plant as of July 1, 1989. Select plant changes made after that cutoff date that could have a significant impact on the model have been incorporated. A review of plant changes from the cutoff date up to July 1, 1992 was completed prior to the submittal of the IPE report; none of these changes are expected to have a major impact on the results. Other PRA studies were also reviewed: NUREG-1150 for Zion and Sequoyah, and the Crystal River 3 PRA of 1987.

Licensee personnel were involved in all aspects of the analysis and contributed more than 50% of the total effort. The licensee was performing almost all the analysis in the latter half of the project (except internal flooding analysis). The contractor was SAIC with ERIN Engineering performing the flooding analysis.

The analysis was reviewed at three levels. ERIN Engineering provided outside review. Plant personnel were also involved in a formal review, as well as an ongoing review as part of the QA procedures.

Waterford 3 PSA staff were involved with the collection of data, interviews of operators, and performance and review of the calculations to determine the HRA probabilities. The analysis was initially performed by an expert from SAIC, with Waterford 3 staff assuming progressively more responsibility. All work on the HRA after December 1990 was performed by Waterford staff. A contractor with "a high level" of PSA expertise (ERIN Engineering) provided an external review of all aspects of the IPE, but "the review team did not include an HRA expert." Regarding the IPE HRA representing the as-built, as-operated plant, the submittal states that "the HRA task served as an integral advisor to other project tasks to assure that relevant human interactions were identified and properly incorporated into the logic models." The HRA task was involved during initial sequence and modeling efforts and "during this period had the opportunity to review plant and system design information and become familiar with the control room and related operating procedures." While simulator exercises were not conducted, the statements discussed above suggest that the HRA analyst was significantly involved throughout the modeling effort. Thus, it appears that steps were taken to assure that the HRA represented the as-built, as-operated plant. However, it was not clear that the HRA gave detailed consideration of plant-specific

factors in determining the HEPs. There was no mention of any walkdowns of important or time consuming operator actions. Response times for actions outside the control room were based on interviews with operators. Both pre-initiator actions (performed during maintenance, test, surveillance, etc.) and post-initiator actions (performed as part of the response to an accident) were addressed in the IPE. A list of important human actions (as determined with a Fussell-Vesely analysis) was provided, as was a list of several recommended improvements to plant procedures.

The Back-end containment analysis was performed by the utility with training and assistance from SAIC.

The Waterford 3 IPE process as described in the submittal seems to satisfy the intent of Generic Letter 88-20.

The licensee intends to maintain a living PRA.

E.3 IPE Analysis

E.3.1 Front-End Analysis

The methodology chosen for the front-end analysis was a Level 1 PRA; the small event tree-large fault tree with fault tree linking approach was used. The computer code used for modeling and quantification was CAFTA.

The IPE quantified the following initiating event categories: 3 LOCAs, 16 transients, one SGTR, one ISLOCA and 1 flooding initiator. The IPE developed 7 event trees to model the plant response to these initiating events. The flooding analysis utilized the existing transient event tree.

Success criteria were based on other PRAs, licensing basis analyses and more realistic calculations.

Containment heat removal is needed in recirculation to assure NPSH of core injection pumps. LPSI pumps cannot operate in recirculation together with containment spray pumps due to NPSH concerns.

The RCP seal LOCA model assumes LOCA occurs only if the operators fail to trip the RCPs within 30 minutes of a loss of CCW.

The data collection process period was 1985 to 1989, with the EDG data period extended to 1991. Plant specific component failure data were only used for EDGs; all other components use generic data. Plant specific data were used exclusively for unavailabilities due to test and maintenance activities.

Waterford 3 data are generally consistent with the NUREG/CR-4550 data. The TDEFW run failure data is substantially lower, and MDEFW CCF factors are somewhat lower. The LOOP and small LOCA initiating event frequencies appear low. The power recovery curve is substantially lower (up to an order of magnitude) than that used in NSAC-147.

The beta factor approach was used for common cause failures, using established procedures. For some components, MGL approach was used.

The internal core damage frequency is 1.7E-5/yr. Of this, flooding contributes 1.1E-6/yr. The internal accident types and initiating events that contribute most to the CDF and their percent contributions are listed below in Tables E-1 and E-2:

Table E-1 Accident Types and Their Contribution to the CDF

Initiating Event Group	Contribution to CDF (/yr)	%
Transients	8.69E-6	51.9
LOCAs	6.62E-6	39.5
Internal Flooding (not included in TOTAL)	(1.12-6)	(6.7)
Steam Generator Tube Rupture	8.26E-7	4.9
Interfacing Systems LOCA	4.86E-7	2.9
ATWS	1.30E-7	0.8
TOTAL INTERNAL CDF	1.68E-5	100.0

Table E-2. Dominant Initiating Events and Their Contribution to the CDF

Initiating Event	Contribution to CDF (/yr)	%
Loss of Offsite Power	7.58E-6	45.3
Small LOCA	5.30E-6	31.6
Medium LOCA	1.14E-6	6.8
Steam Generator Tube Rupture	8.26E-7	4.9
Feedline Break Upstream of Feedwater Isolation Valves	5.18E-7	3.1
ISLOCA (V event)	4.86E-7	2.9
Loss of Feedwater	3.91E-7	2.3
Large LOCA	1.82E-7	1.1
ATWS	1.30E-7	0.8

E.3.2 Human Reliability Analysis

The HRA process for the Waterford 3 IPE addressed both pre-initiator actions (performed during maintenance, test, surveillance, etc.) and post-initiator actions (performed as part of the response to an

accident). The analysis of pre-initiator actions included both miscalibrations and restoration faults. A screening analysis was performed and pre-initiator human actions surviving screening were quantified in more detail using the "SAIC method" described in the book *Human Reliability Analysis* by Dougherty and Fragola. Post-initiator human actions modeled essentially included both response-type (rule-based) and recovery-type actions, but the terminology and categorization was somewhat different. For the post-initiator screening analysis, the modeled sequences were first quantified considering only four top logic post-initiator operator actions. After initial quantification, surviving cutsets were examined and appropriate post-initiator operator actions were added. These actions, including in- and ex-control room actions were quantified using a time reliability correlation approach developed by SAIC and documented in the book by Dougherty and Fragola and in an American Nuclear Society conference paper by Dougherty (1989). In the response to the RAI, the basic form of the TRC is provided along with discussions regarding the relevant input parameters for both an in-control room model and an ex-control model (i.e., for actions to be performed outside the control room). Brief discussions of the input parameters were also provided in the submittal. The critical elements for the in-control room model include: the available response time and an estimate of the median response time for the event examined, along with adjustments for type of behavior (verification, rule-based, and response type, see section 2.3.2.1 for descriptions), degree of "crew burden", success likelihood (an index that can be used to reflect the impact of PSFs), and model uncertainty. For the ex-control room model, similar parameters are modeled, along with adjustments to response time for potential "delaying hazards" outside the control room. The model uncertainty factor can also be adjusted for uncertainty due to other influences or hazards. Hazard factors which can influence response time include lighting, instrument separation, need for tools, need for protective clothing, and other miscellaneous hazards.

One potential limitation of the post-initiator analysis concerns the extent to which plant-specific factors were considered. While the model itself provides reasonable mechanisms for addressing relevant plant -specific factors, on the basis of examples provided, it would appear that many of the parameters were left at their default values and that potential PSFs were not carefully considered. The resulting analysis therefore appears to be "generic" rather than plant-specific and may or may not adequately represent the plant. At a minimum, judgments were made regarding the extent to which operators are burdened in particular scenarios and the type of task involved.

Consideration of dependencies between separate tasks was essentially treated by assuming they are independent. The licensee argues that "between separate tasks independence is provided because many of the tasks are performed by different people, and there is separation in time or "cognitive space", i.e., cues are independent enough to force subsequent diagnosis." The licensee further states "that context effects were handled by lumping the different sequences into one event." "This is done by using a sum average time for the available time parameter for events that are sequence dependent." These statements apparently reflect a "bounding" approach that could lead to pessimistic or optimistic HEPs, depending on the circumstances. A list of important human actions was provided and it was noted that several improvements to plant procedures were recommended. A list of the improvements are provided in section 2.6 of this report.

E.3.3 Back-End Analysis

The Approach Used for Back-End Analysis

Plant Damage States (PDSs) are used as the initial conditions for the Level 2 analysis. The PDSs are defined in the IPE by an event tree structure with the parameters that are important to Level 2 accident progression as the top events. Quantification of accident progression involves the development of a small containment event tree (CET) with the top events of the CET determined by logic trees (i.e., fault trees). The CET and its supporting logic trees developed in the IPE address all the containment failure modes discussed in NUREG-1335.

Quantification of the CET and its supporting logic trees is based on the review of industry literature, primarily the NUREG-1150 document, and plant-specific analyses using the MAAP code. In general, the quantification process for the CET and the associated logic trees is systematic and traceable. The results of the CET analyses lead to an extensive number of CET end states, which are binned into 12 containment release categories (CRCs). Release fractions for the CRCs are calculated in the Waterford 3 IPE by a method similar to that developed in the NUREG-1150 analyses (i.e., the parametric XSOR code).

For the Waterford 3 IPE, the PDS definition scheme is reasonable. The CET is well structured and easy to understand. The CET quantification is also systematic and traceable. The IPE process is in general logical and consistent with GL 88-20.

Back-End Analysis Results

Except for SBO and bypass PDSs, the PDSs defined in the IPE are based on RCS pressure, which depends on the type of accident sequences (or initiators), the time of core melt, which depends on whether core cooling is lost during the injection or the recirculation phase, and the availability of containment systems. The most probable PDS obtained in the Waterford 3 IPE is a PDS with medium RCS pressure (made up primarily by small LOCA sequences), early core melt, and failure of containment heat removal (21% CDF). This is followed by a SBO PDS with early core melt (21%), a SBO with late core melt (17%), and a transient PDS with early core melt, but with containment heat removal available (15% CDF).

Table E-3 shows the probabilities of containment failure modes for Waterford 3 as percentages of the total CDF. Results from the NUREG-1150 analyses for Surry and Zion are also presented for comparison.

Two sets of data are presented in Table E-3 for Waterford 3. The data presented in the original IPE submittal are based on an overly conservative containment fragility curve. This leads to a very high conditional probability of early containment failure. A revised containment fragility curve, which is more consistent with that used in other IPEs, leads to significantly lower early containment failure probability.

Table E-3 Containment Failure as a Percentage of Total CDF

Containment Failure Mode	Waterford 3 IPE+	Waterford 3 IPE Update++	Surry NUREG-1150	Zion NUREG-1150
Early Failure	26	4	0.7	1.4
Late Failure	20	25	5.9	24.0
Bypass	8	8	12.2	0.7
Isolation Failure	***	***	*	**
Intact	46	63	81.2	73.0
CDF (1/ry)	1.7E-5	1.7E-5	4.0E-5	3.4E-4

+ The data pr esented for Waterford 3 are based on Table 4.8-1 of the IPE submittal.
++ Data presented in this column are those obtained from using a revised containment fragility curve (reported in the response to a follow-up RAI).
* Included in Early Failure, approximately 0.02%
** Included in Early Failure, approximately 0.5%
*** Included in Early Failure, approximately 0.1%

Of the 8% conditional probability of containment bypass failure presented in the above table, 5% comes from SGTR sequences and 3% comes from ISLOCA sequences. The contribution from ISGTR is negligible. The effect of restarting the RCPs on ISGTR, which is considered in some other IPEs as a mechanism that may increase the potential of induced SGTR, is not considered important in the Waterford 3 IPE.

The conditional probability of early containment failure presented in the original IPE submittal is about 26% of total CDF. The major threat to early containment failure is a combination of the loss of containment heat removal with the RCS is at high pressure. For this case, the containment is at elevated pressure due to steam generation such that a high pressure melt ejection (HPME) can challenge containment integrity. This occurs during SBO sequences, or in small LOCA sequences with the loss of both safety injection and containment heat removal (CHR). Of the 26% early failure probability, over 13% is from SBO sequences and over 11% is from small LOCA sequences. On a conditional basis, about 35% of SBO sequences result in early failure and 30% of small LOCA sequences result in early failure. According to the licensee's response to RAI follow-up questions, although the probability of early containment failure is significantly reduced by the use of a revised containment fragility curve (from 26% to 4%), the dominant sequences that lead to early containment failure remain the same as that described in the IPE submittal.

The conditional probability of late containment failure presented in the IPE submittal is 20%. The major contributor to late containment failure is steam overpressurization when CHR is lost. SBO does not contribute as much to late containment failures because of the high likelihood of AC power recovery (before containment failure). Of the 20% late failure probability about 15% is from small LOCA, 4% from SBO, and 1.3% from other transients. On a conditional basis, about 39% of small LOCA sequences, 12% of large LOCA sequences, 10% of SBO sequences, and 9% of other transients result in late failure. According to the licensee's response to the RAI, the conditional probability of late containment failure increased from 20% to 25% when the revised containment fragility is used. Since

detailed data are not provided in the RAI responses, contributions from the various accident sequences to late containment failure cannot be obtained. It seems that the increase in late containment failure probability is primarily due to the decrease of early containment failure probability, and the dominant sequences that lead to late containment failure remain the same as that described in the IPE submittal.

Source terms for the containment release categories (i.e., the CET end states) are determined by a method similar to that used in NUREG-1150 studies. Source terms are presented in the IPE submittal in terms of release fractions for noble gases, Iodine, Cesium, Tellurium, and Strontium. Except for the SGTR release category, the release fractions obtained in the Waterford 3 IPE for the various release categories seem to be consistent with those obtained in other IPEs. For the SGTR release category, the release fractions obtained in the IPE are based on the availability of water scrubbing. Since water scrubbing may not be available for all SGTR sequences and the release fractions for the SGTR sequences without water scrubbing may be much greater than those with water scrubbing, the release fractions reported in the submittal for SGTR sequences may not be adequate for some SGTR sequences. Although the omission of the source term for SGTR sequences without water scrubbing is not a significant problem in the present IPE because of their small frequency in comparison with those of other sequences that have large releases (e.g., ISLOCA), it is a deficiency nonetheless. It would be desirable to divide the SGTR CRC to two CRCs with and without water scrubbing and to obtain the source terms for both of them. This would assure that significant information is not lost in the IPE process in the future IPE update.

Two types of sensitivity studies are performed in the IPE to determine key assumptions on the final results. The first type of sensitivity studies are probabilistic in nature and address uncertainties in the quantification of the various containment failure modes modeled in the CET. The second type of sensitivity studies involve deterministic analyses using the MAAP code, performed in the IPE to ensure that a broad spectrum of possible outcome are covered in the IPE. The issues investigated in the sensitivity studies of the first type include ex-vessel cooling, RCS depressurization due to hot leg creep rupture, ultimate containment pressure, reactor cavity wall structure failure, frequency of two important PDSs, hydrogen combustion, DCH, and debris bed coolability. The parameters investigated in the sensitivity studies of the second type include in-vessel hydrogen production, DCH, debris coolability, and vessel penetration radius. The sensitivity studies provided in the Waterford 3 IPE seems to have addressed the issues of significant uncertainties in the IPE analysis.

E.4 Generic Issues and Containment Performance Improvements

The IPE addresses decay heat removal (DHR). CDF contributions were estimated for the following DHR methods: secondary cooling (main feedwater, auxiliary feedwater, emergency feedwater, condensate, turbine bypass and atmospheric dump valves) and primary inventory control (HPSI and charging systems). Failures of the EFW and HPSI were found to make a major contribution to the total CDF. The EFW failures in the most important sequences are dominated by TDEFW pump failure to start, MDEFW pump common cause failures, operator failure to provide EFW suction when CSP is exhausted. The HPSI failures are caused by common cause of A and B pumps, operator or mechanical failures with pump AB, failure in the CCW system to provide HPSI pump cooling and HVAC failures.

The DHR function contributes less than the 3.0E-5/yr criterion for the "acceptably low" DHR contribution in NUREG-1289. Therefore, this issue is considered closed.

No other generic issues are discussed in the submittal.

The CPI recommendation for PWRs with a dry containment is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements. Although the effects of hydrogen combustion on containment integrity and equipment are discussed in the submittal, the CPI issue is not specifically addressed in the submittal. More detailed information on this issue is provided in the licensee's response to the RAI. According to the response, although no containment walkdowns were conducted specifically for Level 2, the Waterford 3 PSA staff has made many trips to the containment and has a good understanding of the geometry of the containment.

According to the response, the Waterford 3 containment is a very open design that is not compartmentalized, and with the possible exception of the reactor cavity, all parts of the containment atmosphere are expected to be well mixed during an accident scenario. The reactor cavity is the only relatively enclosed volume in the containment. Since the reactor cavity volume is surrounded by thick reinforced concrete walls sized to withstand a large break LOCA blowdown and since no equipment is located in this area, hydrogen combustion in the cavity is not expected to affect any safety significant equipment. Additionally, according to the response, hydrogen detonation is not believed to be likely in the Waterford 3 containment. As can be seen in the above description, the discussions provided by the licensee on this issue is qualitative in nature, no quantitative information is provided in the discussion.

E.5 Vulnerabilities and Plant Improvements

The licensee defined a vulnerability as either an extremely high sequence CDF (substantially greater than 1.E-4/yr), a greater than 50% contribution to CDF from a single sequence or an event that contributes in an unusual or substantial way to the risk profile. No vulnerabilities were found.

No credit for plant improvements was given in the IPE. The following proposed improvements will be resolved as part of the severe accident guidance framework, to be completed by summer of 1997 (guidance for using LPSI for CSS recirculation has already been implemented:

Hardware:

1) Install a portable generator to charge the AB battery. This will reduce SBO contribution from depletion of this battery which is used to control the TDEFW pump.

2) Provide feedwater from the fire protection system to the steam generator. The fire protection system has its own diesel driven pumps. During SBO or total loss of feedwater, this system could be used provided the SG were depressurized to below 200 psia, the shutoff head of these pumps.

Operating procedures:

1) Provide additional chiller/HVAC failure guidance. Room cooling is important as a contributor to the CDF and because it cools HPSI and EFW (MD) pumps. The failures are typically slow acting so the operators have time to respond. Therefore additional guidance may insure a timely response.

2) Cross-tie of AC power trains. Proceduralize the cross-tie between the A and B trains (hardware already exists). Drills have demonstrated the pertinence of this type of recovery. A procedure will make it easier to accomplish it in a shorter time.

3) Enhance refill of the CSP. CSP drawdown is an important contributor. Emphasizing the need to monitor level and makeup from the wet cooling tower basins or the CST will help prevent this from being a contributor.

4) Add guidance for aligning LPSI pump for containment spray. Containment cooling is needed in the recirculation phase to insure NPSH of recirculation pumps. Hardware connections exist for LPSI to take over the recirculation spray function in case of CSS pump failure, however, currently, LPSI pumps are disabled from recirculation. This is because they would cavitate if operated together with the CSS pumps to take suction from the containment sump. Since in this case CSS pumps are not available, LPSI pumps can take over to provide CHR. This procedure guidance has already been implemented.

No CDF change from these improvements has been estimated.

E.6 Observations

Based on the Level 1 review of the Waterford 3 IPE the licensee appears to have analyzed the design and operations of Waterford 3 to discover instances of particular vulnerability to core damage. It also appears that the licensee has: developed an overall appreciation of severe accident behavior; gained an understanding of the most likely severe accidents at Waterford 3; and implemented changes to the plant to help prevent and mitigate severe accidents. It is not clear that quantitative understanding was gained by the licensee due to a number of data problems (see below).

Strengths of the Level 1 IPE are as follows: Thorough analysis of initiating events and their impact, descriptions of the plant responses, modeling of accident scenarios, generally reasonable failure data and common cause factors employed and usage of plant specific data where possible to support the quantification of initiating events, diesel generator failures and component maintenance unavailabilities. The flooding analysis seems to have been reasonable and thorough. The effort seems to have been evenly distributed across the various areas of the analysis. The documentation was usually good, and reasonable effort was made to provide RAI responses. Some pessimistic assumptions were employed to offset some of the optimistic aspects of the analysis.

The weaknesses of the IPE were the following:

- using seemingly low values for some important initiator frequencies (LOOP and small LOCA)
- offsite power recovery curve is very optimistic
- omission of some component classes from common cause analysis (air compressors, relays, switches, check valves, fans, etc.)
- omission of the third HPSI pump, the third CCW pump and the third chiller from the common cause analysis on the basis of different operating regimes from the other two trains in the system.

- HVAC modeling of the shutdown heat exchanger room seems to be optimistic

- shedding of DC loads in station blackout is not modeled

- TDEFW pump run failure rate is low (2 orders of magnitude) compared to the NUREG-1150 recommended value (but is in line with some other IPEs and apparently some generic data sources).

There were some aspects of the analyses which may have offset some of the weaknesses: EDG run failures occur at the beginning of the SBO, no credit for TDEFW pump operation with water at inlet, large maintenance unavailability of the dry cooling tower, no credit for recent battery upgrades such that load shedding may not be required.

The IPE determined that failures in the AC power, EFW, ACCW, HPSI, CCW and HVAC dominate the risk profile. Loss of offsite power and small LOCA account for about 80% of the total CDF. SBO accounts for about 38% of the CDF. The CDF is dominated by 5 accident sequences (not accounting the ISLOCA which contributes about 3%).

The HRA review of the Waterford 3 IPE submittal and a review of the licensees responses to HRA related questions asked in the NRC RAI, revealed several weaknesses in the HRA as documented. In general, a viable approach (the Dougherty and Fragola method) was used in performing the HRA, but several weaknesses in how the analysis was conducted (or at least in the licensees documentation of the conduct of the analysis) were identified. While the weaknesses are not severe enough to conclude that the licensees submittal failed to meet the intent of Generic Letter 88-20 in regards to the HRA, they do suggest the licensee may not have learned as much about the role of humans during accidents as would have been possible. Important elements pertinent to this determination include the following:

1) The submittal indicates that utility personnel were significantly involved in the HRA. Regarding the IPE HRA representing the as-built, as-operated plant, the submittal states that "the HRA task served as an integral advisor to other project tasks to assure that relevant human interactions were identified and properly incorporated into the logic models." The HRA task was involved during initial sequence and modeling efforts and "during this period had the opportunity to review plant and system design information and become familiar with the control room and related operating procedures." While simulator exercises were not conducted, the statements discussed above suggest that the HRA analyst was significantly involved throughout the modeling effort. Thus, it appears that steps were taken to assure that the HRA represented the as-built, as-operated plant. However, documentation of HRA related walkdowns and observations of simulator exercises would have strengthened the notion that a viable process was used.

2) The submittal indicted that the analysis of pre-initiator actions included both miscalibrations and restoration faults. An acceptable, but potentially optimistic analysis was conducted. Events found to be potentially risk significant were analyzed in detail using an "SAIC" method that is "a variant on THERP and is similar to the ASEP HRA procedure.

3) The major limitation of the post-initiator analysis concerns the extent to which plant-specific factors were considered. While the model itself provides reasonable mechanisms for addressing relevant plant - specific factors, on the basis of examples provided, it would appear that many of the parameters were left at their default values and that potential PSFs were not carefully

considered. The resulting analysis therefore appears to be "generic" rather than plant-specific and may or may not adequately represent the plant.

4) Consideration of dependencies between separate tasks was essentially treated by assuming they are independent. The licensee argues that "between separate tasks independence is provided because many of the tasks are performed by different people, and there is separation in time or "cognitive space", i.e., cues are independent enough to force subsequent diagnosis." The licensee further states "that context effects were handled by lumping the different sequences into one event." "This is done by using a sum average time for the available time parameter for events that are sequence dependent." These statements apparently reflect a "bounding" approach that could lead to pessimistic or optimistic HEPs, depending on the circumstances.

5) A list of important human actions based on their contribution to core damage frequency was provided in the submittal.

6) The HRA portion of the flooding analysis appeared reasonable and thorough.

The following are the major findings of the back-end analysis described in the submittal:

- The back-end portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20.
- The Waterford 3 Steam Electric Station IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.
- The IPE has identified a plant-specific reactor cavity configuration feature that may affect accident progression. Based on the IPE, it is recommended that the communication between sump and cavity be enhanced. This may be achieved by removing the door in the cavity cooling ductwork to increase the flow of the water in the containment sump to the reactor cavity.
- The containment analyses indicate that there is a 46% conditional probability of containment failure. The conditional probability of containment failure is about 8% for containment bypass, 26% for early containment failure, and 20% for late containment failure.
- The high early containment failure probability obtained in the IPE submittal (26%) is primarily due to the use of a conservative containment failure probability curve (or containment fragility curve). The early failure probability is reduced to 4% if a containment failure probability curve consistent with that used in other IPEs is used.
- The CPI issue is not addressed specifically in the IPE submittal. It is discussed in the licensee's response to one RAI questions. However, the response is qualitative in nature.

NOMENCLATURE

ACCW	Auxiliary Component Cooling Water
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
AHU	Air Handling Units
ALWR	Advanced Light Water Reactor
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BHEP	Basic Human Error Probability
BNL	Brookhaven National Laboratory
CCF	Common Cause Failure
CCW	Component Cooling Water
CDF	Core Damage Frequency
CE	Combustion Engineering
CET	Containment Event Tree
CHR	Containment Heat Removal
CPI	Containment Peformance Improvement
CRC	Containment Release Category
CS	Containment Spray
CSS	Containment Spray System
CST	Condensate Storage Tank
DHR	Decay Heat Removal
EDG	Emergency Diesel Generator
EFAS	Emergency Feedwater Actuation System
EFW	Emergency Feedwater
EOS	Equipment Out-of-Service
GL	Generic Letter
HEP	Human Error Probability
HFE	Human Failure
HPME	High Pressure Melt Ejection
HPSI	High Pressure Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilating and Air Conditioning
IPE	Individual Plant Examination
ISGTR	Induced SGTR
ISLOCA	Interfacing Systems LOCA
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPSI	Low Pressure Safety Injection
MAAP	Modular Accident Analysis Package
MDEFW	Motor Driven EFW
MFW	Main Feedwater
MGL	Multiple Greek Letter

NOMENCLATURE (Cont'd)

NPSH	Net Positive Suction Head
NRPDS	Nuclear Plant Reliability Data System
PDS	Plant Damage State
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWSP	Refueling Water Storage Pool
SAIC	Science Applications International Company
SBO	Station Blackout
SDC	Shutdown Cooling
SGTR	Steam Generator Tube Rupture
SIAS	Safety Injection Actuation System
SIMS	Station Information Management System
SLI	Success Likelihood Indeces
SUPS	Static Uninterruptible Power Supplies
SUT	Startup Transformer
TDEFW	Turbine Driven EFW
TER	Technical Evaluation Report
THERP	Technique for Human Error Rate Prediction
TRC	Time Reliability Correlations
UAT	Unit Auxiliary Transformer
W3	Waterford 3

1 INTRODUCTION

1.1 Review Process

This technical evaluation report (TER) documents the results of the BNL review of the Waterford 3 Steam Electric Station Individual Plant Examination (IPE) submittal [IPE submittal, RAI Responses]. This technical evaluation report adopts the NRC review objectives, which include the following:

- To assess if the IPE submittal meets the intent of Generic Letter 88-20, and
- To determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335.

A Request of Additional Information (RAI), which resulted from a preliminary review of the IPE submittal, was prepared by BNL and discussed with the NRC. Based on this discussion, the NRC staff submitted an RAI to Entergy Operations, Inc. on January 22, 1996. Entergy Operations, Inc. responded to the RAI in a document dated April 30, 1996, and to follow-up questions in a document dated August 29, 1996 (RAI Responses). This TER is based on the original submittal and the responses to the RAIs.

1.2 Plant Characterization

The Waterford 3 Steam Electric Station is a 1153 MWe, 3410 MWth Combustion Engineering pressurized water reactor (PWR). The reactor coolant system (RCS) consists of the reactor vessel, two U-tube steam generators, 4 shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnected piping. The plant is operated by Entergy Operations, Inc., and started commercial operation in the Fall of 1985. There are no other operating units on site.

Design features at Waterford 3 that impact the core damage frequency (CDF) are as follows:

- There is no feed and bleed capability at this plant. No pressurizer PORV exists and the HPSI/charging pumps do not have the requisite head to lift the safety valves.
- The turbine driven main feedwater pumps will continue to run for most transients, as the pump flow output is automatically matched to the decay heat level.
- There are two motor driven (capacity 350 gpm each) and one turbine driven (capacity 700 gpm) EFW pump. The EFW system is automatically started and controlled. In addition, a manually started AFW pump is also available, should the other three pumps fail (the AFW pump is normally used during startup/shutdown operations). According to the submittal and the RAI responses, the turbine driven EFW pump can be expected to continue to operate with low quality steam or even water at the turbine inlet. However, this is not credited in the analysis, and the TDEFW pump is assumed failed at the time of battery depletion.
- The normal EFW suction source is the inventory in the condensate storage pool (CSP), good for about 10 hours. A backup supply are the two wet cooling tower basins, each holding about the same amount of water as the CSP. A third option is the non-seismically qualified condensate storage tank (CST) and its transfer pump.

- The EFW control valves fail open on loss of instrument air, and there is also a backup nitrogen accumulator supply in case of loss of instrument air. The turbine driven EFW pump does not require room cooling (according to calculations, RAI responses), whereas the motor driven EFW pumps do.

- The DC battery (battery AB) supplying control to the TDEFW pump has a SBO depletion time of 4 hours with proceduralized load shedding (1 hour without load shedding), according to the submittal. Since the IPE, the safety related batteries have been replaced with higher capacity batteries (to allow for aging), and a new non-safety battery has been installed to take up the non-safety loads serviced by the AB battery. These modifications have extended the AB battery depletion time to 6 hours.

- Condensate pumps may be used to provide feedwater to the steam generators, provided the secondary system has been depressurized to 500 psia. There are three parallel condensate pumps. The condenser hotwells have enough inventory to supply the condensate pumps for 24 hours.

- There are multiple pathways for secondary steam relief: 6 turbine bypass valves, 2 atmospheric dump valves and 6 safety relief valves.

- The RCP seals are the Byron Jackson type, which according to the submittal can sustain loss of CCW for 30 minutes (verified by tests), without tripping the RCPs; the operators are instructed to trip the RCPs immediately upon loss of CCW. CCW cooling is the only type of cooling for these seals (no seal injection provided). Because of the 4 stage seal design, and the new resistant material for seal faces, no spurious seal failures (i.e., initiating event seal LOCA) are assumed possible with these seals (consequential failures are allowed).

- There are three trains of HPSI, CCW, AC safety buses and DC safety buses. The AB buses and AB trains are functionally related, e.g., the AB train of CCW cools the AB train of HPSI, and both are supplied AC power from the AB safety bus. The third HPSI pump must be manually started on SI.

- There are also three trains of HVAC chillers. The charging pumps also have three trains (these are considered in the PRA analysis to feed the auxiliary pressurizer spray, for emergency boration in ATWS and for RCS inventory control in an SGTR). The other safety equipment has two trains. The two trains of the instrument air compressors are backed up by the three trains of the station air compressors (see below).

- There are two EDGs. The EDGs need cooling by CCW, ventilation by dedicated fans and DC power provided by the station batteries. A diesel compressor has been added to the plant post-IPE, to help in case of problems with startup compressed air.

- There are three plant batteries, A, B, and AB. The AB battery is used for TDEFW pump control in SBO conditions. As stated above, the capacity of this battery has been increased and a non-safety battery added to pick up non-safety AB loads, such that SBO depletion time of this battery is now 6 hours. The A and B batteries have also been similarly affected, such that their SBO depletion time is now 4 hours even without load shedding. Each battery is supported by two chargers.

- There is no service water system at this plant. Instead, the ultimate heat sink is provided by the dry cooling towers. As there are multiple fans in the towers, they can be maintained piecemeal, such that maintenance would not disable the whole tower (although in the IPE it is conservatively assumed that it does). Also, in case of increased demand (depending on air temperature) and during normal operation there are additional wet cooling towers which are used to increase the heat rejection capacity. The IPE assumes that the wet cooling towers are needed in case of a LOCA, when several types of safety equipment may be operating simultaneously. The system which cools the CCW system and rejects the heat to the wet cooling towers is known as the auxiliary component cooling system (ACCW), and is only needed in case of LOCAs, as far as the IPE is concerned. This system has two pump trains and two wet cooling towers.

- The CCW is needed to cool the HPSI pumps, the LPSI pumps, containment spray pumps, shutdown heat exchangers (also used for containment spray recirculation cooling), containment fans, the emergency diesel generators and the central chillers used to provide HVAC cooling for several plant areas.

- The instrument air system is necessary for operation of the MFW system and the normal pressurizer spray (but not the auxiliary spray, supplied by the charging pump). All the other important systems (EFW, CCW, ACCW, containment sump recirculation valves) are provided with a backup air or nitrogen accumulator system. There are two instrument air compressors, of which one is sufficient to supply the requisite loads in an intermittent type of operation. In case of failure of both compressors, a cross tie to the station air system automatically opens; the station air has three compressors. Therefore the compressed air system seems to be relatively reliable and the systems affected are relatively few.

- Room cooling or ventilation is needed for several important systems: HPSI (not needed during the RWSP injection phase due to the low temperature of the water pumped), LPSI (not needed during the injection phase), containment sprays (not needed in the injection phase), MDEFW pumps, normal pressurizer sprays, emergency diesel generators and the CCW pumps.

- The switchover to recirculation is automatic. However, the operator must manually close the RWSP (refueling water storage pool) suction valves at that time.

- The recirculation spray (using the CSS pumps aligned to the containment sump and the shutdown heat exchangers) is necessary to provide cooling of the containment sump water.

- LPSI is automatically stopped on switchover to recirculation and HPSI is automatically aligned to the sump (along with the CSS) even if LPSI operated in the injection mode, and even though a LPSI path for recirculation (through the shutdown heat exchangers) exists. The reason is that the LPSI pumps may cavitate when simultaneously taking suction from the containment sump with the containment spray pumps. Since the IPE, a hardware modification has been implemented such that the LPSI pumps can be used to provide the recirculation spray in case of failure of the spray pumps.

The Waterford 3 Steam Electric Station utilizes a large dry containment consisting of a freestanding steel vessel surrounded by a reinforced concrete shield building. Some of the plant characteristics important to the back-end analysis are summarized in Table 1 of this report.

Table 1 Plant and Containment Characteristics for Waterford 3 Steam Electric Station

Characteristic	Waterford 3	Zion	Surry
Thermal Power, MW(t)	3390	3236	2441
RCS Water Volume, ft^5	11,100	12,700	9200
Containment Free volume, ft^3	2,680,000	2,860,000	1,800,000
Mass of Fuel, lbm	223,900	216,000	175,000
Mass of Zircalloy, lbm	64,100	44,500	36,200
Containment Design Pressure, psig	44	47	45
Median Containment Failure Pressure, psig	135	135	126
RCS Water Volume/Power, ft^3/MW(t)	3.3	3.9	3.8
Containment Volume/Power, ft^3/MW(t)	791	884	737
Zr Mass/Containment Volume, lbm/ ft^3	0.024	0.016	0.020
Fuel Mass/Containment Volume, lbm/ ft^3	0.084	0.076	0.097

Both the thermal power level and the containment free volume of Waterford 3 are similar to those of Zion. With the exception of the mass of Zircalloy in the reactor system (and thus its ratio to containment volume), the values of other parameters are also similar to those of Zion. It is noted that the parameters presented in the above table provide only rough indications of the containment's capability to meet severe accident challenges and that both the containment strength and the challenges associated with the severe accident involve significant uncertainties.

The plant characteristics important to the back-end analysis are:

- A cavity design which facilitates flooding of the reactor cavity. According to the IPE, water can readily flow from the containment sump to the reactor cavity. Flooding of the cavity is accomplished through a small tunnel that connects to the ductwork that provides reactor cavity cooling. Flooding of the reactor cavity and the low placement of the reactor vessel in the reactor cavity ensures that ex-vessel cooling can occur.

- A steel shell containment that is vulnerable to direct attack by dispersed core debris. However, based on the consideration of potential debris dispersing paths and MAAP calculations, the Waterford IPE discounts the possibility of direct corium attack on the steel containment wall.

- A reactor vessel with no lower head penetrations. This delays the time of vessel failure, but may cause a more energetic failure with larger hole size.

- The larger amount of Zircalloy in the core assemblies. The amount of Zircalloy in the core assemblies of Waterford 3 is about 40% more than that of Zion. The amount of hydrogen produced during a severe accident is thus more for Waterford 3 than for Zion.

- A small reactor cavity with very little area for ejected core material to disperse to the upper containment region. The cavity is open to the upper compartment through a very small annulus between the vessel and cavity wall.

- The large containment volume, high containment pressure capability, and the open nature of compartments which facilitates good atmospheric mixing.

2 TECHNICAL REVIEW

2.1 Licensee's IPE Process

2.1.1 Completeness and Methodology

The licensee has provided the type of information requested by Generic Letter 88-20 and NUREG 1335.

The front-end portion of the IPE is a Level 1 PRA. The specific technique used for the Level 1 PRA was a small event tree/large fault tree, with fault tree linking and it is clearly described in the submittal.

Internal initiating event and internal flooding were considered. Event trees were developed for all classes of initiating events. Several sensitivity analyses were performed (all basic events with Fussell-Vesely importance of at least 1% had their failure rate/probability increased by an order of magnitude). Importance (F-V) of basic events was calculated. System importance analysis was also performed.

The submittal information on the HRA process was generally inadequate in scope. Additional information/clarification was obtained from the licensee through an NRC request for additional information. The HRA process for the Waterford 3 IPE addressed both pre-initiator actions (performed during maintenance, test, surveillance, etc.) and post-initiator actions (performed as part of the response to an accident). The analysis of pre-initiator actions included both miscalibrations and restoration faults. A screening analysis was performed and pre-initiator human actions surviving screening were quantified in more detail using the "SAIC method" described in the book *Human Reliability Analysis* by Dougherty and Fragola. The post-initiator human actions modeled essentially included both response-type (rule-based) and recovery-type actions, but the terminology and categorization was somewhat different. For the post-initiator screening analysis, the modeled sequences were first quantified considering only four top logic post-initiator operator actions. After initial quantification, surviving cutsets were examined and appropriate post-initiator operator actions were added. These actions, including in- and ex-control room actions were quantified using time reliability correlation approach developed by SAIC and documented in the book by Dougherty and Fragola and in an American Nuclear Society conference paper by Dougherty (1989). In the response to the RAI, the basic form of the TRC is provided along with discussions regarding the relevant input parameters for both an in-control room model and an ex-control model (i.e., for actions to be performed outside the control room). Brief discussions of the input parameters were also provided in the submittal. The critical elements for the in-control room model include: the available response time and an estimate of the median response time for the event examined, along with adjustments for type of behavior (verification, rule-based, and response type, see section 2.3.2.1 for descriptions), degree of "crew burden", success likelihood (an index that can be used to reflect the impact of PSFs), and model uncertainty. For the ex-control room model, similar parameters are modeled, along with adjustments to response time for potential "delaying hazards" outside the control room. The model uncertainty factor can also be adjusted for uncertainty due to other influences or hazards. Hazard factors which can influence response time include lighting, instrument separation, need for tools, need for protective clothing, and other miscellaneous hazards.

One potential limitation of the post-initiator analysis concerns the extent to which plant-specific factors were considered. While the model itself provides reasonable mechanisms for addressing relevant plant -specific factors, on the basis of examples provided, it would appear that many of the parameters were left at their default values and that potential PSFs were not carefully considered. The resulting analysis

therefore appears to be "generic" rather than plant-specific and may or may not adequately represent the plant.. At a minimum, analysts had to make judgments were made regarding the extent to which operators are burdened in a particular scenarios and the type of task involved.

Consideration of dependencies between separate tasks was essentially treated by assuming that they are independent. The licensee argues that "between separate tasks independence is provided because many of the tasks are performed by different people, and there is separation in time or "cognitive space", i.e., cues are independent enough to force subsequent diagnosis." The licensee further states "that context effects were handled by lumping the different sequences into one event." "This is done by using a sum average time for the available time parameter for events that are sequence dependent." These statements apparently reflect a "bounding" approach that could lead to pessimistic or optimistic HEPs, depending on the circumstances. A list of important human actions was provided and it was noted that several improvements to plant procedures were recommended. A list of the improvements are provided in section 2.6 of this report.

The Waterford 3 Steam Electric Station Individual Plant Examination (IPE) back-end submittal is essentially consistent with respect to the level of detail requested in NUREG-1335. The methodology employed in the Waterford 3 IPE for the Level 2 evaluation is clearly described in the submittal. Plant Damage States (PDSs), which are defined in the IPE by an event tree structure with the parameters important to Level 2 accident progression as the top events, are used as the initial conditions for the Level 2 analysis. Quantification of the Level 2 accident progression involves the development of small top level containment event trees (CETs). The top events of the CETs are determined by the fault trees (called logic trees in the IPE submittal). The CETs and the supporting logic trees addressed in detail all the containment failure modes discussed in NUREG-1335. The results of the CET analyses are an extensive number of CET end states which are binned into twelve containment release categories (CRCs). The CET quantification relies on review of industry literature, primarily the NUREG-1150 document, and plant-specific analyses using MAAP code. Release fractions for the CRCs are calculated in the Waterford 3 IPE by a method similar to that developed in the NUREG-1150 analyses (i.e., the parametric XSOR code).

The IPE was initiated in late 1988. The model reflects the plant as of July 1, 1989. Select plant changes made after that cutoff date that could have a significant impact on the model have been incorporated. A review of plant changes from the cutoff date up to July 1, 1992 was completed prior to the submittal of the IPE report; none of these changes are expected to have a major impact on the results. Other PRA studies were also reviewed: NUREG-1150 for Zion and Sequoyah, and the Crystal River 3 PRA of 1987.

2.1.2 Multi-Unit Effects and As-Built, As-Operated Status

There are no other operating units on site.

A wide variety of up-to-date information sources were used to develop the IPE: FSAR system description, piping and instrumentation drawings, electrical one line drawings, system design basis documents, licensee event reports, monthly operating reports, technical specifications, emergency operating procedures and special studies and analyses. The analysis was applied to the plant configuration as it existed in mid 1989. The data was collected from September 24, 1985 to March 31, 1989 (for maintenance data); the data window was extended to December 31, 1991 for the emergency diesel generator failure data. Other components use generic data. Walkdowns were performed if there were

questions with a specific aspect of modeling (also there were frequent interactions with persons familiar with various aspects of the plant). Due to the newness of the plant, it is expected that the plant documentation, drawings, etc. accurately represent the as built as operated plant (RAI responses). In addition, a flooding analysis walkdown was performed.

The submittal states that "the HRA task served as an integral advisor to other project tasks to assure that relevant human interactions were identified and properly incorporated into the logic models." The HRA task was involved during initial sequence and modeling efforts and "during this period had the opportunity to review plant and system design information and become familiar with the control room and related operating procedures." While simulator exercises were not conducted, the statements discussed above suggest that the HRA analyst was significantly involved throughout the modeling effort. Thus, it appears that steps were taken to assure that the HRA represented the as-built, as-operated plant. However, it was not clear that the HRA gave detailed consideration of plant-specific factors in determining the HEPs. There was no mention of any walkdowns of important or time consuming operator actions. Response times for actions outside the control room were based interviews with operators. No human related multiunit effects were identified.

Insofar as the back-end analyses are concerned, it appears that all the Waterford 3 containment specific features are modeled.

It seems the licensee intends to maintain a "living PRA".

2.1.3 Licensee Participation and Peer Review

The licensee contributed "well over 50% of the total engineering effort (about nine man-years) applied to the project". The licensee contracted with SAIC to develop the PRA and transfer the technology to Entergy personnel. SAIC was on board from the start of the PRA in September of 1988 until the PRA development contract expired in December of 1990. Thereafter, Waterford 3 personnel had sole responsibility for all aspects of the PRA. Initially, the relationship consisted mostly of learning and assistance by utility engineers. Waterford 3 personnel and SAIC shared the initial development of system fault trees, quantification and evaluation. As the project progressed, utility involvement and expertise in all aspects of the PRA increased. Since December of 1990, all analytical work, including additional development of plant models for both Level 1 and Level 2, data analysis, quantification, and evaluation of results have been performed by Waterford 3 personnel with minor assistance from outside firms. The internal flood analysis is the single exception since the bulk of the technical work there was performed by ERIN Engineering personnel with Waterford 3 personnel input and assistance. Entergy staff participated, particularly during the data collection and plant walkdown, and result review phases of the project. Utility engineers were involved in assuring that all the components in affected flood areas were accounted for and that the Level 1 basic events representing those components were appropriately tagged. The Entergy staff were involved in directing the contractor on key assumptions and operator recovery actions that could be credited. Finally, the same staff reviewed and approved the final results of the analysis to ensure a clear understanding of the analysis details and results by the utility.

The reviews performed for the IPE included both independent in-house reviews and an external review. There were three levels of review: normal engineering quality assurance carried out by the organization performing the analysis, which consisted of a qualified individual with knowledge of PRA methods and plant systems performing an independent review of all assumptions, calculations and results for each task

and system model in the Level 1 analysis (except the internal flood analysis). The second level of review was performed by plant personnel not directly involved with the development of the PRA model and consisted of individuals from Operations, Engineering, Training and Licensing groups who reviewed the system models and accident sequence description. The third level of review was performed by PRA experts from ERIN Engineering. This review was conducted in two phases. During the first phase, the review team concentrated on the overall PRA methodology, accident sequence analysis and system fault trees. The intent was to provide early feedback to the Waterford 3 staff concerning the adequacy and accuracy of the reviewed products. The second phase included Level 1 results, human failure and recovery analysis, preliminary plant damage state cutsets and a preliminary CET (Level 2). The intent of this phase was to identify any modeling inaccuracies, inappropriate failure data, inconsistencies between cut sets, reasonableness of recoveries and results, and making sure the cut sets were properly binned into the PDSs. A summary of the major areas of review comments is provided in the submittal. In addition to the above review, a review was performed by experts from ABB Combustion Engineering for the Level 2 analysis.

A slight concern is that the utility did not continue with the original contractor through the end of the analysis. Even though the intent was to transfer knowledge, some continuity may be lost due to imperfect transfer of memory of nuances of the analysis, assumptions, justification for such assumptions, etc. Also the flooding analysis contractor was different than the original Level 1 contractor. This may be partially offset by hiring an outside contractor to do an early review as the project was progressing.

Another area of concern is that there was apparently no outside independent review of the flooding analysis, as ERIN Engineering both performed the flooding analysis and was involved in the Level 1 review work.

The PRA team for the Waterford 3 IPE consisted of Waterford 3 Design Engineering and "Corporate" Engineering personnel. This was supplemented by Science Application International Corporation (SAIC) and other outside consultant firms experienced in PRA methods and applications. The Waterford 3 personnel had sole responsibility for the PRA model after December 1990, when the contract with SAIC expired.

From the description provided in the IPE submittal it seems that the intent of Generic Letter 88-20 is satisfied.

2.2 Front End Technical Review

2.2.1 Accident Sequence Delineation and System Analysis

2.2.1.1 Initiating Events

The identification of initiating events proceeded in a two-stage approach: 1) review of existing sources, including other PRAs of similar plants (Calvert Cliffs, ANO-2 and Crystal River 3), EPRI documents (EPRI NP-2230 and NSAC-152), and the NRC accident sequence precursor reports (NUREG/CR-3591 and NUREG/CR-4674), and, 2) a thorough review of each frontline and support system at Waterford 3 to identify failures that could lead to an initiating event.

As a result, a total of 22 initiating events (including 1 flood) were identified. In addition, the reactor vessel rupture was not mentioned in the submittal (as part of the 22 initiators), but was later discussed in the RAI responses. The internal initiators are:

LOCAs:

Large LOCA
Medium LOCA
Small LOCA

Transients:

Reactor trip
Loss of condenser vacuum
Turbine trip
Loss of feedwater
Loss of offsite power
Steamline break
Feedline break upstream of the main feedwater isolation valve
Loss of condensate system
Loss of component cooling water system
Loss of 6.9 kV bus 3A1
Loss of 6.9 kV bus 3B1
Loss of 125 V DC bus 3A
Loss of 125 V DC bus 3B
Loss of 125 V DC bus 3AB
Loss of power distribution panel 3014-AB
Loss of instrument air

Other events:

Steam generator tube rupture
Interfacing LOCA, suction line to shutdown cooling system
Internal flood in turbine building

The initiating event list seems to be mostly complete and comparable to events considered in other PRAs.

HVAC failures do not lead to initiating events because of a low probability of failure of all three chillers, long time scales to reach damaging temperatures and availability of DHR equipment which would not be affected by HVAC failures (turbine driven EFW pump, the AFW pump and the condensate pumps).

As stated above, spurious failure of RCP seals was not considered a credible initiator (this is usually considered a very small LOCA), due to the nature of the Byron Jackson seals.

It is not clear why a loss of a 4.2 kV bus was not considered as an initiator (as opposed to the 6.9 kV bus). Unlike a loss of the 6.9 kV buses, a loss of a non-safety 4.2 kV bus would also cause a loss of the associated 4.2 kV safety bus. Furthermore, a non-safety 4.2 kV bus is "required" for normal plant operation. The RAI responses just reiterate (without explaining) that a loss of this bus would not cause an automatic plant trip due to unspecified "redundancies" in design of the non-safety power system. In

any case, based on the reviewer's experience with other PRAs, usually a loss of a 4.2 kV bus is not expected to have a major impact on the CDF.

A failure of the pressurizer pressure control system was not included as an initiator due to existence of control room alarms and perceived low conditional core damage probability.

Reactor vessel rupture was not included for probabilistic reasons, but in response to the RAIs it was stated that WASH-1400 analysis would be accepted. This should result in a pessimistic evaluation of this initiator as the Waterford 3 vessel is made with more fracture resistant materials than older vessels and there are no in-core detector lower head penetrations.

2.2.1.2 Event Trees

The IPE developed 7 event trees: the general transient event tree, the station blackout event tree, the ATWS event tree, the small LOCA event tree, the medium LOCA event tree, the large LOCA event tree and the SGTR event tree. No event tree was developed for the interfacing LOCA, however it was assumed that once the isolation failure in the SDC suction occurred (initiating event), there was a 50% chance that there would be a pipe rupture outside the containment, with core melt and containment bypass resulting (pipe ruptures inside the containment would add insignificantly to the existing LOCA frequencies). Existing event trees were used for the flooding analysis (a general transient event tree used for the surviving scenario in the turbine building, which causes a loss of offsite power).

The event trees are functional. The mission time used in the core damage analysis was 24 hours, unless shorter time was indicated (e.g., LOCA injection phase).

The event tree end states are divided into two possible outcomes: success or core damage.

It appears the analysts used core uncovery as the definition of core damage for most initiators, along with the limit on clad temperature for larger LOCAs.

Success criteria are based on review of other PRAs (ANO-2, Crystal River, Zion, etc.), licensing accident analyses presented in the FSAR and more realistic accident analyses previously performed for Waterford 3. The success criteria appear reasonable and in line with most other PWR success criteria.

Large LOCAs require injection from all three safety injection tanks attached to the intact loops, and injection from 1/2 LPSI pumps into one intact loop, and injection from 1/3 HPSI pumps into at least 2/3 intact loops.

For the recirculation phase of all three LOCAs, containment heat removal via either recirculation sprays or the fan coolers is required.

Small and medium LOCAs require control rod insertion for reactivity control. Both small and medium LOCAs require HPSI pump injection from RWSP (only large LOCA requires LPSI injection from RWSP).

Recirculation for all LOCAs is accomplished by the use of HPSI pumps; LPSI pumps are not used (even though the hardware setup exists) due to NPSH problems when taking suction from the containment sump

in conjunction with operation of the containment spray pumps in recirculation mode (RAI responses). HPSI pumps can be aligned to either hot leg or cold leg recirculation.

In case of small LOCAs, heat removal through one of the steam generators is also required, as the break flow is insufficient to remove all decay heat. This can be accomplished by the use of one MFW pump, or one EFW pump, or the AFW pump, or one condensate pump in conjunction with secondary system depressurization (using 1 atmospheric dump valve or one steam bypass to the condenser valve). This same heat removal criterion is applied to transients, LOSP and long term ATWS heat removal.

The pressure control success criterion for ATWS specifies 3700 psia as the limiting RCS pressure. This is considered conservative as CE analyses indicate a failure pressure of 4300 psia. The CE analyses encompassed stress evaluations of all major primary and auxiliary RCS components within the CE purview. It was concluded that peak pressures of up to 4300 psia would not jeopardize the integrity or the operability of equipment needed for safe shutdown. The 3700 psia success criterion is below the pressure at which CE analyses show that the upper reactor vessel head would lift to relieve pressure. Also, CE tests with severely wasted steam generator tubes show that consequential SGTR would not occur at these pressures.

Early operation of the EFW system will help in limiting peak pressures, depending on the moderator temperature coefficient and operation of the turbine trip function. A potential for common cause failure between the RPS and the emergency feedwater actuation system (EFAS) was conservatively modeled; this does not take into account the existence of a diverse EFAS which has no such commonality.

A turbine trip helps to minimize the RCS pressure by maximizing the available inventory in the steam generators. It is pessimistically assumed that turbine trip will fail if ATWS was due to an electrical failure. Also, no credit was taken for turbine trip as an initiating event (although it was included in the ATWS initiating event frequency).

Long term reactivity control via emergency boration is modeled by operation of (two) charging pumps taking suction from a boric acid makeup tank; no credit for RWSP suction is given.

Both pressurizer safety valves have to open in ATWS for successful pressure control.

Pressure control requirements are more stringent for transients and SGTR sequences than at other PWRs, due to a lack of a feed and bleed capability. In transients, the operator is required to isolate the pressurizer heaters and isolate the RCS makeup, while secondary steam relief via turbine bypass valves or ADVs is also necessary for certain transients. In case of an SGTR the operator is required to throttle the HPSI flow, in addition to performing the RCS depressurization. Credit is given to operation of 2/3 charging pumps for inventory control in SGTR (as an alternative to 1/2 pumps) if the RCS is depressurized sufficiently such that the charging pumps' lower flow rate can provide adequate makeup. An assumption is made that the operators will not wait for the automatic reactor trip on low RCS pressure (15 minutes). A late trip causes a reactor upper head void to grow upon depressurization, possibly interfering with natural circulation.

It is shown that there is enough inventory in the RWSP to last beyond the mission time even assuming failure to isolate the faulted steam generator. Nevertheless, failures beyond the 24 hour time frame were included in the recovery analysis (failure to initiate the SDC, failure to refill the RWSP, etc.).

As stated above, the RCP seals are the Byron Jackson type. It is assumed they would fail only if the operators fail to trip the RCPs within 30 minutes of a loss of CCW.

Success criteria for CCW assume that in case of LOCAs the wet cooling tower cooling would be needed (via ACCW) to supplement the dry cooling tower(s) due to an increased heat load. In case of other accidents, either dry cooling tower cooling or wet cooling tower cooling can supply the required UHS (the minimum number of fans operating in either type of the cooling tower is specified in the success criteria for all accidents).

The station blackout tree pessimistically assumes that the diesel generator failure to run occurs at the start of the sequence, i.e., no allowance is given to longer core uncovery times later in the accident. This is offset by non-modeling of DC load shedding to preserve the batteries, as claimed in the submittal. The latter non-conservatism has been ameliorated recently with installation of new batteries of higher capacity and adding a non-safety battery, such that depletion times have been increased since the IPE, and load shedding apparently does not need to be modeled for the depletion times assumed in the model.

The SBO tree also takes credit for the fact that failure of TDEFW pump to run (before the expiration of the 4 hr battery depletion time) can occur any time between 0 and 4 hours; thus, in case of failure to run, an average running time of 2 hours is assumed, which gives an extended time for core uncovery (1.5 hours vs. 50 minutes at the start of the accident).

2.2.1.3 Systems Analysis

A total of 15 systems/functions are described in Appendix B of the Submittal. Included are descriptions of the following systems: AC power, component cooling water, containment spray, DC power, emergency feedwater, engineered safety features actuation, high pressure safety injection, instrument air and station air, low pressure safety injection, power conversion and main feedwater, pressurizer pressure control, room cooling (HVAC), safety injection tanks, containment cooling (fan coolers) and containment isolation system. In addition, the RAI responses contained a more detailed and helpful description of the HVAC system and its modeling.

Each system description includes a discussion of the system design and operation, details of modeling and assumptions, system interfaces (support systems), test and maintenance requirements, success criteria and system level initiators.

Also included for many systems are simplified schematics that show major equipment items and important flow and configuration information.

System dependencies are summarized in a matrix form.

Section 1.2 of this TER describes the important plant features.

2.2.1.4 System Dependencies

The IPE addressed and considered the following types of dependencies: shared component, instrumentation and control, isolation, motive power, direct equipment cooling, areas requiring HVAC, and operator actions. There is not much discussion of environmental effects, apart from HVAC and

flooding/spray considerations. The effect of the flood on cable terminal points, such as junction boxes, was also considered.

In case of HVAC, RAI responses provided a detailed description of HVAC design, rooms requiring HVAC and HVAC modeling considerations. The HVAC consists of the three main chillers (cooled by the CCW system) and the three chiller pumps, providing chilled water to the air handling units in individual rooms/plant areas. Some areas just have ventilation fans, i.e., there are no chilled air handling units (e.g., EDG rooms).

The following key rooms have chilled AHUs (all rooms except for safeguard pump AB room contain two AHUs; the safeguard pump AB room contains only one AHU):

1) control room;
2) control room (mechanical equipment room);
3) switchgear area, cable vault and battery rooms;
4) CCW heat exchangers (heat transfer from CCW to ACCW);
5) CCW pump AB;
6) CCW pumps;
7) switchgear area
8) safeguard pump AB (contains HPSI pump AB);
9) safeguard pumps A (HPSI, LPSI, CSS pump A);
10) safeguard pumps B;
11) shutdown heat exchangers;
12) emergency feedwater pumps;
13) charging pumps;
14) charging pump AB.

The important areas served by fans only are the HVAC equipment room, turbine building switchgear room and emergency diesel generators rooms.

Not all the above rooms require room cooling, however. The following areas do not require room cooling:

1) TDEFW pump area (reason: SBO heatup calculations, the room is actually a cage in a large area;

2) Battery rooms A, B and AB (reason: temperature calculations for SBO conditions, hydrogen purge function judged unnecessary during an accident);

3) Safeguards pump rooms, during injection of RWSP water only (due to low temperature of RWSP water), required in recirculation phase;

4) CCW heat exchanger rooms A and B (reason: valves tested by manufacturer to well over 300°F, with only speed of opening/closing affected, CCW temperature is on the order of 100°F or lower);

5) Control room (reason: shutdown of plant in 1 hour if both trains inoperable, slow heatup, existence of remote shutdown panel);

6) Shutdown heat exchanger room (reason: equipment assumed not affected by high room temperature);

7) HVAC equipment room (reason: large room, containing relatively few heat loads, cooled by CCW, ventilation fans are assumed not required);

8) Turbine building switchgear room (reason: large area, three walls are outside walls, slow heatup, served by ventilation fans only).

There seems to have been a relatively complete consideration of HVAC in the model. In case of item 6), shutdown heat exchangers, there does not seem to be enough justification for not considering HVAC failures. This would primarily impact LOCA recirculation sequences.

Table 3.2-5 of the submittal contain the overall system dependency matrix, including both support-on-support and frontline-on-support dependencies.

2.2.2 Quantitative Process

2.2.2.1 Quantification of Accident Sequence Frequencies

The IPE used a small event tree/large fault tree technique to quantify core damage sequences. The event trees were functional. The CAFTA workstation software package was used for development and quantification of top event probabilities and accident frequencies.

It appears the cut set truncation limit used was 2.E-9/yr. The truncated residuals are a negligible fraction of the CDFs, according to the submittal.

The IPE took credit for various recovery activities, including the recovery of offsite power. The only diesel recovery modeled are simple recoveries in the air supply for starting the diesels, but this is not significant, according to the submittal. The IPE power recovery curve is given by the equation:

$$P_{nonrecov} = \exp(-0.88 * t^{0.89}),$$

where the time t is given in hours (RAI responses). No reference is given for this equation. Table 2 shows a comparison between the offsite nonrecovery probability at the times of interest calculated by the above equation and that given in NSAC-147. The latter EPRI document contains industry average data on offsite power recovery.

Table 2 IPE vs. NSAC-147, Nonrecovery of Offsite Power

Time after initiator (hr)	IPE probability of nonrecovery of offsite power	NSAC-147 probability of nonrecovery of offsite power
1	0.41	0.46
3.5	0.068	0.17
6	0.013	0.12
8	3.7E-3	0.08

The times of interest are mostly from the SBO event tree considerations: core uncovery occurs at approximately 1 hour after the SBO initiator if the TDEFW pump does not start. If the TDEFW pump starts but fails during its run time, an average core uncovery time of 3.5 hours after the initiator is calculated. With battery depletion time of 4 hours and 2 additional hours for core uncovery, 6 hours is calculated in case the TDEFW pump runs until the battery discharges Finally, 8 hours is given for comparison on how the results diverge at longer times. It can be seen that at 3.5 hours, the IPE is optimistic by a factor of 2.5. At 6 hours, the underestimation of core uncovery probability is almost one order of magnitude. This may have a significant impact on the results. For example, one of the dominant sequences is SBOVL (frequency of 2.86E-6/yr, or 17.1% of the present CDF). This is a station blackout, with battery depletion after 4 hours and non-recovery of offsite power in 6 hours. In addition, the LOOP nonrecovery within 6 hours has an overall Fussell-Vesely importance of 0.186. Raising this probability by an order of magnitude would mean raising the total CDF by almost a factor of 3 and the above sequence would then be almost 60% of the new CDF.

It appears that, in comparison to NSAC-147 data, the offsite power recovery factors are very optimistic and will significantly impact the results.

2.2.2.2 Point Estimates and Uncertainty/Sensitivity Analyses

Mean values were used for the point estimate initiator frequencies and all other basic events. No uncertainty analysis was performed on the results. Importance measures (Fusell-Vesely) are given for systems, basic events, initiating events, and sequences. The most important basic event are the following, each one having a F-V importance > 5%: LOOP nonrecovery within 50 minutes, failure of TDEFW pump to start, LOOP nonrecovery within 6 hours, EDG failures to run, operator failure to recover from room cooling failures, common cause failure of the EDGs to run, dry cooling tower maintenance unavailabilities, common cause failure of the containment sump recirculation valves, EDG start failures and ACCW pump failures.

Sensitivity studies were also performed. For each basic event whose F-V importance was greater than 1%, failure probabilities were arbitrarily and individually increased by an order of magnitude. Certain classes of events were also increased by an order of magnitude (all CCF failures, etc.). The following were components and classes of events to which the CDF was most sensitive and the IPE calculated change in CDF: all motor driven pump failures (610% change in CDF), all common cause failures (278%), all test and maintenance (264%), all operator recovery errors (176%), all pre-event human errors (146%), all MOV failures (115%); EDG fails to start and fails to run (increased failure rates by the error factors, 508%), EDG fails to run (438%), MDEFW pumps fail to start (228%), LOOP nonrecovery in 50 min (increased $P_{nonrecov}$ to 1.0) and 6 hours (186%), EDG failure to start (173%), ACCW pump failure to start (155%), various EDG demand failures (142%), dry cooling tower unavailability due to maintenance (111%) and failure to restore AHUs in the switchgear areas (111%).

2.2.2.3 Use of Plant Specific Data

Since the plant is relatively new and there hasn't been enough time to develop plant specific data, mostly generic data was used, except for the maintenance data and the diesel generator failure data.

The data collection process period was from September 24, 1985 to March 31, 1989 (for maintenance data); the data window was extended to December 31, 1991 for the emergency diesel generator failure data.

For maintenance and test data, the data sources examined were the computerized records of SIMS (station information management system), NPRDS (nuclear plant reliability data system) and the control room EOS (equipment out of service) log.

While the test and maintenance data appear reasonable, it was noted that the error factors presented were too small. The licensee agrees and states that the wrong method was used to estimate. There is no impact on the results. The unavailability for the dry cooling towers was pessimistically estimated such that whenever there was maintenance on a tower fan, the whole multi-fan cooling tower is declared out of service. In reality sections of the tower can be separated from the rest. This affects the LOCA sequences.

The data for the diesel generators was taken from the technical specification surveillance program which mandates a certain number of EDG tests. There have been 3 start failures in 228 demands and one run failure in 412 hours of operation. The plant specific EDG data is then obtained by straight division of the appropriate numbers (no Bayesian updating was used).

Table 3 of this review compares the failure data for selected components from the IPE to values typically used in PRA and IPE studies, using the NUREG/CR-4550 data for comparison [NUREG/CR 4550, Methodology]. Most of the data in the table is generic data.

Waterford 3 data are generally in agreement with the NUREG/CR-4550 data. The data for circuit breaker failure to transfer and EDG failure to start data is somewhat lower than that used in NUREG/CR-4550. The turbine driven pump failure to run data is significantly lower. In RAI responses, the licensee points out that this is an aggregate of 4 generic data sources, all of which have a much small run failure rate of TD pumps than does NUREG/CR-4550. Also, comparison is made to demonstrated reliability of the TD MFW pumps which have to run continuously. The licensee states that the high 4550 values are due to data from Peach Bottom which was an outlier. It is not clear if the TDEFW pump should have the same data as the TDMFW pump, and what is the right value to use for the TDEFW pumps (most other IPEs seem to go with the lower value). The data used for the TD EFW pump can have some impact on the results. If the 4550 data are used for failure to run, it is expected that the SBOVL sequence contribution would increase by a multiplier of 1.8 (an 80% increase) while the total CDF would increase by about 13% (reviewer's estimates).

2.2.2.4 Use of Generic Data

As discussed in Section 2.2.2.3 above, most failure data used in the IPE and presented in Table 3 were actually generic data. The data mostly comes from the SAIC generic data base. Sometimes data was aggregated from several sources using the SAIC CARP program. For example in case of turbine driven pumps data from the following sources were aggregated: ASEP data base, NUREG/CR-2886 (IPRD), NUREG/CR-1205 and the NREP data base.

Table 3 Comparison of Failure Data

Component	Failure Mode	Waterford 3 Data	4550 Data
Turbine driven pump	fail to start	2.6E-2	3.0E-2
	fail to run	8.9E-5/hr	5.0E-3/hr
Motor driven pump	fail to start	4.8E-3	3.0E-3
	fail to run	8.4E-5/hr	3.0E-5/hr
Instrument air compressor	fail to start	1.3E-1	8.0E-2
	fail to run	2.5E-3/hr	2.0E-4/hr
Battery charger	fails to operate	7.8E-6/hr	1.0E-6/hr
Circuit breaker	spurious open	1.9E-6/hr	1.0E-6/hr
	fail to transfer	1.2E-3	3.0E-3
AC bus	fault	1.2E-7/hr	1.0E-7/hr
Check valve (AFW)	fail to open	1.4E-4	1.0E-4
	fail to close	1.6E-3	1.0E-3
MOV	fail to close/open	5.5E-3	3.0E-3
	spurious open	1.4E-6/hr	5.0E-7/hr
	spurious close	1.5E-6/hr	1.0E-7/hr
Emergency diesel generator (the only plant specific data in Table)	fail to start	1.3E-2	3.0E-2
	fail to run	2.4E-3/hr	2.0E-3/hr

Notes: (1) 4550 are mean values taken from NUREG/CR-4550, i.e. from the NUREG-1150 study of five U.S. nuclear power plants.

(2) Demand failures are probabilities per demand. Failures to run or operate are frequencies expressed in number of failures per hour.

2.2.2.5 Common-Cause Quantification

The common cause probabilities were based on the procedure presented in NUREG/CR-4780 and the data presented in EPRI NP-3967. It seems that the approach used was the beta factor approach, for most components. The submittal states that no credible data exists to support common cause failure analysis of components other than pumps, MOVs, EDGs and batteries (although EDG ventilation fans and chillers are also included in the data). Chiller B was assumed not to have CCF with chillers A and AB due to a different mode of operation: chiller B operates continuously, while chillers A and AB alternate monthly, therefore the same wear would not be experienced. This is not a very compelling argument (chiller failure is a relatively important contributor to the CDF). The check valve CCFs were neglected probabilistically (check valves have lower failure probabilities than components they are in series with, e.g. MOVs, pumps). Only EDG fans CCF was modeled (CCF factors for other fans were not available at the time of the IPE); the other HVAC fans will have lower common cause failure rates than the associated chillers. The licensee states that there is no evidence in the data reviewed for Waterford 3 of CCFs for circuit breakers (other than reactor trip breakers), electrical switchgear, air operated valves, air compressors, inverters, relays, transmitters (except miscalibration which is modeled in the HRA) (RAI responses). Other PRAs have included these failures.

CCF of pressurizer safety valves was not considered because failure of only one valve in the stuck open mode would cause a small LOCA. Likewise, in an ATWS, failure of one safety valve to open would be a failure of the pressure control function.

The common cause failure between the TDEFW pump and the two MDEFW pumps was not considered credible due to a different driver (the pump parts are similar, though the TDEFW pump is somewhat larger). This is not expected to have a major influence on the results or the conclusions of the study (RAI responses). The licensee also states that the AFW pump is of a totally different design and thus not subject to CCF with the EFW pumps. However, there could be common problems such as steam binding. Also note that the MDEFW pump CCF is by about a factor of 2 lower than the 4550 recommended values. EFW is an important system, contributing about 30% to core damage, thus it is important that it be modeled correctly. Failures of pumps with different drivers (common cause) have been modeled in PRAs and are included in EPRI documents (for example the ALWR requirements data base).

It is not clear if common cause failure of all three HPSI pumps or all three CCW pumps was considered (no CCF factors are provided and this does not appear in dominant sequences). If not, this may have a significant impact on the results.

A comparison of effective β factors in the submittal vs. those suggested in NUREG/CR-4550 ("reference β factor") is presented in Table 4.

The table shows general consistency between the Waterford 3 CCF data and that recommended in NUREG/CR-4550. Most of the CCF factors are in agreement, except the MDEFW pumps' and the CSS pumps' CCF factors are lower by a factor of 2 in the IPE. The MDEFW CCF may have some measurable impact on the results.

In conclusion, the CCF analysis, while mostly reasonable, may have had a measurable effect on the results in the direction of understating the contributions to the CDF at Waterford 3.

Table 4. Comparison of Common-Cause Failure Factors

Component	Submittal β factor	Reference β factor
HPSI pumps, LPSI pumps	0.17	0.21 HPSI 0.15 LPSI
Containment spray pumps	0.05	0.11
Chiller pumps, MDEFW pumps, CCW pumps, ACCW pumps	0.03	0.026(SW)0.056 EFW
Chillers	0.11	
MOV, CCF of 2 valves	0.08	0.088
Battery	0.05	
EDG ventilation fans	0.13	
Diesel Generator, CCF of 2 EDGs	0.05	0.038

2.2.2.6 Initiating Event Frequency Quantification

The initiating event frequencies were calculated by three methods: Waterford 3 specific experience, generic industry data (but specified to Waterford specific design) and fault tree modeling of Waterford 3 systems using generic industry data. No Bayesian updating was used.

The plant specific experience was used for the reactor trip, turbine trip, loss of feedwater. Plant specific fault trees with generic failure data were used for loss of instrument air and loss of PDP 3014-AB. Generic data were used for all other initiators; for loss of offsite power, the generic data were adjusted for plant specific features.

The initiating event frequencies used in the IPE are presented in Table 5.

The initiating event frequencies generally seem reasonable and are comparable to other PRA studies. The large LOCA, small LOCA and LOOP frequencies seem lower than expected.

The large LOCA employs a leak before break consideration which reduces it to an order of magnitude below that of NUREG/CR-4550 recommendations. Other studies are quoted which have this frequency at 1.E-4/yr before leak before break considerations. An estimate from NUREG/CR-4290 is quoted for a large guillotine break at CE plants of 5.E-14/yr. In any case this is not expected to have a large impact on the CDF results as large LOCA now presents about 1% of the total CDF.

The small LOCA frequency is lower than the NUREG/CR small LOCA (1.E-3/yr) plus very small LOCA (1.3E-2/yr) combination, by about a factor of 3. This frequency was calculated by dividing the two applicable industry events to this category by the total number of PWR years. The two events assigned to the small LOCA category were at Robinson on 3/7/71 and Zion on 12/31/73. There are no PORV contributions at this plant (no pressurizer PORVs), and the RCP seals are judged to be sturdy enough such that spurious RCP seal LOCA is not deemed credible.

The ANO-1 RCP failure event (along with 4 others quoted in NUREG/CR-4550) is dismissed as inapplicable, due to improvements in seals or a different seal design. It is also stated that the ANO-1 event was not a LOCA as the leak rate was within the charging pump makeup capacity ("according to the ANO engineers"). Also, the pressure never fell to the SIAS setpoint, according to the RAI responses. However, NUREG/CR-4550 (Vol. 3, Rev. 1, Part 2, App. D.2) quotes a leak rate of 400 gpm for that event (which the RAI responses states was an overestimate), with a total spillage of 60,000 gallons. This event, which occurred in 1980, plus the one at Oconee 2 in 1974 (leak rate 90 gpm, total leakage 50,000 gallons) seem to have occurred in Byron Jackson RCP seals, and both were spurious failures (not caused by loss of seal cooling/injection), according to NUREG/CR-4550.

It is possible that there have been substantial improvements in seal design and materials since that time. However, NUREG/CR-4550 accounts for that by doing a Bayesian updating of early failures in the period 1974 through 1980 with lack of failures in the period 1981-1988. This yields an estimate of 3.9E-3/yr for spurious RCP seal LOCAs.

Also, other categories of very small LOCAs are estimated in NUREG/CR-4550 to have a frequency of 1.7E-3/yr for very small LOCA pipe breaks and 7.6E-3/yr for component boundary failures. Events with leakage rates greater than 15 gpm were counted, which occurred during startup or power operation, and,

for pipe breaks, ratio of LOCA sensitive piping to all other piping of 18% for Westinghouse plants was used.

It is true that the effects of a spurious RCP seal LOCA (usually considered a very small LOCA) would probably be bounded by the small LOCA accident sequence logic (i.e., the conditional core damage probability of a small LOCA is probably larger than that of a very small LOCA). Since the small LOCA frequency at Waterford 3 of 4.5E-3/yr is a few times larger than that recommended in NUREG/CR-4550 (1.0E-3/yr), and considering lack of PORVs and improvements in RCP seal design, the total impact of small LOCAs on the plant risk profile is probably not severely underestimated (from the standpoint of initiator frequency). There is probably some underestimation, based on the state of our knowledge of these phenomena.

The LOOP frequency of 0.032/yr seems somewhat low, but is in line with most other IPEs encountered by this reviewer. This is calculated by culling from the generic data base events deemed inapplicable to the plant, based on the switchyard design, weather patterns, etc.

The licensee states that the generic data base includes events such as ice storms, inapplicable to the site (RAI responses). It is stated that generic data base also includes hurricanes in the North East, and there would probably be a precautionary shutdown in the event of a hurricane, which is usually slow moving (RAI responses). It is stated that tornadoes should be less frequent than in the Midwest, and the responses disagree with the RAI's characterization of the site as being subject to "severe weather relatively frequently". The switchyard design includes two switchyard buses, fed by seven transmission lines, feeding two separate startup transformers, which then feed the 4.2kV safety and non-safety buses. No LOSP events on site have occurred, but there have been 5 partial LOSP events.

It seems the methodology used for specializing the industry occurrence data to the plant specific conditions, tends to underestimate the LOOP frequency. The data appears to include shutdown time in the calculation of the total reactor years, and there may be cases (as in plant centered LOOP) where inappropriate reactor years are not adequately screened out. In any case, the LOOP frequency should not rise by more than a factor of two, which would translate into a correspondigly higher LOOP caused CDF. Since LOOP is already a primary CDF contributor, the conclusion as to LOOP significance to the CDF won't change, but the numerical values will. Also, the observation above on the power nonrecovery factors (they seem low) would bias the LOOP CDF contribution in the same direction as the relatively low LOOP frequency.

In addition, some of the error factors quoted for the initiating events seem low, and sometimes do not make sense when compared on a relative basis (some relatively rare events have a smaller error factors than some relatively frequent events). For instance, the error factor for loss of offsite power is 1.33 (meaning that this number is known with a high degree of certainty), while the error factor for a turbine trip, derived from plant specific occurrences, is 5.20. However, this should have no impact on the results as uncertainty analysis was not performed.

Table 5 Initiating Event Frequencies for Waterford 3 IPE

Initiating Event	Frequency (/yr)
Reactor Trip	2.6
Loss of Condenser Vacuum	0.14
Turbine Trip	0.40
Loss of Feedwater	1.0
Loss of Offsite Power	3.16E-2
Steamline break on SG2	5.60E-3
Feedwater Line Break Upstream of the Feedwater Isolation Valves	5.60E-3
Loss of Condensate System	1.00E-2
Loss of CCW/ACCW	5.00E-3
Loss of 6900 V bus	3.94E-4
Loss of DC Bus	3.94E-4
Loss of PDP 3014AB	2.57E-2
Loss of Instrument Air	4.67E-2
Small LOCA	4.47E-3
Medium LOCA	1.00E-3
Large LOCA	5.00E-5
Reactor vessel rupture	2.7E-7
Steam Generator Tube Rupture	8.94E-3
Interfacing System LOCA (suction valves of shutdown cooling)	9.72E-7
Internal Flooding (Turbine Building, circulating water pipes)	3.05E-3

2.2.3 Interface Issues

2.2.3.1 Front-End and Back-End Interfaces

The IPE assumes that containment heat removal is necessary for core heat removal when recirculation is required. Also, LPSI pumps cannot be used in conjunction with CSS pumps. Both CSS pumps and containment fan coolers require CCW cooling; also CSS pumps require room cooling in recirculation.

Section 2.4 provides more information on Level 2 considerations.

2.2.3.2 Human Factors Interfaces

In case of a fast dead bus transfer (from the auxiliary transformer to a startup transformer) after a plant trip, the failure of this automatic action can be recovered by the operators, either from the control room, or locally, at the breakers. The HEP for the local action is 4.5E-3, the one in the control room (event ZMANTRAN) is on the order of 1.E-6, which seems very low, and they seem to have been assumed independent. This will have an impact on the LOOP frequency, or, for some initiators, the SBO frequency.

Section 2.3 provides more information on HRA considerations.

2.2.4 Internal Flooding

2.2.4.1 Internal Flooding Methodology

The methodology used to perform the flooding analysis consisted of five major steps:

1) Preliminary flood scenario development;
2) Plant walkdown;
3) Initial flood scenario frequency screening;
4) Refinement of analysis bases and assumptions;
5) Detailed quantification of important flood scenarios.

The final two steps were performed iteratively until each scenario was determined to be below the established screening frequency or until the scenario frequency was as low as reasonably achievable using the screening methods of this study. This process may result in a substantial residual not being reported in the final results.

The screening criterion was 1.E-6/yr.

The development of flooding scenarios was supported by a plant walkdown. The effect of the flood on equipment cable terminal points (e.g. junction boxes) was deduced from automated plant cable data bases and Appendix R equipment cable tables obtained from the Waterford 3 design engineering electrical group. Pipe whip and steam impingement were judged as being beyond the scope of the analysis. Liquid jets and sprays were not considered as to the exact patterns of impingement, but were assumed to fail all the equipment in the initiation flood area.

Propagation of flooding to other areas (including open doors, stairwells, elevator shafts, drains, and through gaps in closed doors) and isolation of the floods were considered. Fire doors are not water tight. Sumps are assumed to overflow with sump pumps unable to keep up with the deluge flood flow. Drain plugging was apparently not considered. However, a deluge flood would seek large openings such as stairways and elevator shafts to propagate to lower levels. Isolation of large floods in 20 minutes with a probability of 0.01 is assumed. Inadvertent actuation of the fire suppression equipment and maintenance induced floods are "implicitly part of the data base", i.e., were not separately developed to uncover any plant specific vulnerabilities. Component failures considered which could cause flooding were pipe and valve ruptures, pipe joints, flanges, tanks, etc. Internal flooding data from PLG-0624 of May 1988 were used, and calculation of flood source density in different areas was performed.

In the detailed analysis, minimum water levels to induce equipment damage were considered in the flood propagation zones.

Surviving flood scenarios were quantified using internal events event trees with flood induced failure tagged in the fault trees. Flood revised HEPs were used for recovery actions.

Only one flood scenario survived, the turbine building break in the circulating water system. This causes a loss of offsite power and a loss of the MFW/AFW and condensate systems; the TDEFW fails to start. The result is a demand on the EDGs to power the EFW system located in the reactor aux building. The initiating event frequency for this scenario is 3.05E-3/yr.

It should be noted that scenarios in the reactor aux bldg that propagate to the basement where the EFW equipment is located are screened out due to the large floor areas such that the flood water spreads out and will likely not affect the EFW equipment.

2.2.4.2 Internal Flooding Results

The turbine building flood scenario CDF is 1.12E-6/yr. The residual of the screened scenarios has an upper bound of 3.49E-6/yr (this is estimated using pessimistic screening quantification).

It seems that the flooding analysis was reasonable.

2.2.5 Core Damage Sequence Results

2.2.5.1 Dominant Core Damage Sequences

The results of the IPE analysis are in the form of functional sequences, therefore NUREG-1335 screening criteria for reporting of such sequences are used. The point estimate for the core damage frequency from internal events is 1.68E-5/yr (this does not include a minor contribution from the reactor vessel rupture, 2.7E-7/yr), with internal flooding contributing an additional 1.12E-6/yr. Accident types and their percent contribution to the CDF, are listed in Table 6. The most important initiators are given in Table 7.

Five dominant sequences and one ISLOCA containment bypass sequences were described in detail (two LOCAs, 2 station blackout, 1 general transient, one ISLOCA). Each of these important sequences has a frequency greater than 1.E-6/yr, except ISLOCA, which is greater than 1.E-7/yr. The important sequences are summarized below in Table 8. System Importances are presented in Figure 1.

The RCP seal LOCA contribution is negligible. The SBO contribution is 37%. The LOOP events and the small LOCA are the most important events. This is expected due to the design of the plant (no feed and bleed capability, relatively fast core uncovery, EDG dependencies on CCW, HVAC and DC power, dependence of HPSI on CCW and ACCW, and HVAC, dependencies of MDEFW on HVAC).

Table 6 Accident Types and Their Contribution to the CDF

Initiating Event Group	Contribution to CDF (/yr)	%
Transients	8.69E-6	51.9
LOCAs	6.62E-6	39.5
Internal Flooding (not included in TOTAL)	(1.12-6)	(6.7)
Steam Generator Tube Rupture	8.26E-7	4.9
Interfacing Systems LOCA	4.86E-7	2.9
ATWS	1.30E-7	0.8
TOTAL INTERNAL CDF	1.68E-5	100.0

Table 7 Dominant Initiating Events and Their Contribution to the CDF

Initiating Event	Contribution to CDF (/yr)	%
Loss of Offsite Power	7.58E-6	45.3
Small LOCA	5.30E-6	31.6
Medium LOCA	1.14E-6	6.8
Steam Generator Tube Rupture	8.26E-7	4.9
Feedline Break Upstream of Feedwater Isolation Valves	5.18E-7	3.1
ISLOCA (V event)	4.86E-7	2.9
Loss of Feedwater	3.91E-7	2.3
Large LOCA	1.82E-7	1.1
ATWS	1.30E-7	0.8

Table 8 Dominant Core Damage Sequences

Initiating Event	Dominant Subsequent Failures in Sequence	% of CDF
Small Break LOCA	Safety Injection Failure during injection mode (caused by CCW failure to provide HPSI pump cooling (caused by wet or dry cooling tower problems), or caused by mechanical failures in the A or B HPSI pumps, or operator failure with the AB HPSI pump	23.4
Loss of offsite power (blackout)	failure of both EDGs leading to station blackout, failure of the turbine driven EFW pump (both EDG and TDEFW failures could be due to failures of these components themselves or failures of the DC system, e.g., common cause failure of all three batteries), failure to restore offsite power in 50 minutes	20.2
Loss of offsite power (blackout)	failure of both EDGs leading to station blackout, success of the TDEFW pump, failure to restore offsite power in 4 hours, battery depletion leads to eventual failure of the TDEFW pump	17.1
Transient (dominant are loss of offsite power and feedwater line break upstream of the feedwater isolation valve)	various types of failures disable all three EFW pumps and the AFW pump, in addition to the MFW pumps being disabled by the initiator (feedline break initiator disable both MFW and AFW); the failures are either pump failures (to start, run, common cause, maintenance), or due to problems with suction (problems with condensate storage pool, operator failure to switch to CST and/or the wet cooling tower); failure to depressurize and use condensate pumps	14.6
Small Break LOCA	failure of HPSI to switchover to recirculation (caused by common cause failure of SI sump suction valves, or much less frequently independent failures of the two valves). Much lower in probability are HPSI recirculation failures due to failure of the RWSP low level instruments, plugging of the SI pump and HVAC failures of the HPSI pump room or switchgear room cooling	8.6
High/low pressure boundary failure (ISLOCA initiator, primarily gross failure of the two series MOVs in the shutdown cooling suction line)	low pressure piping failure outside the containment (with a probability of 0.5)	2.9

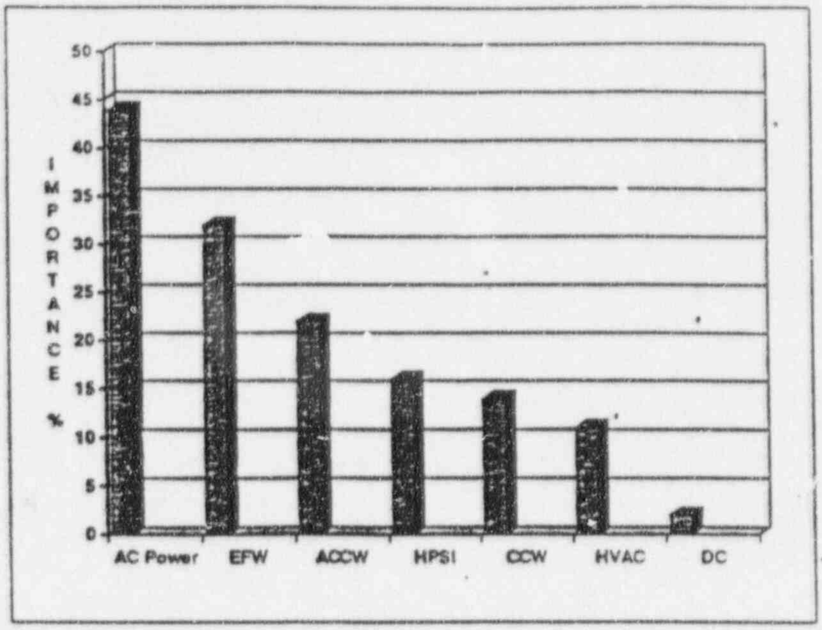

Figure 1 System Importance

2.3 Human Reliability Analysis Technical Review

2.3.1 Pre-Initiator Human Actions

Errors in the performance of pre-initiator human actions (such as failure to restore or properly align equipment after testing or maintenance, or miscalibration of system logic instrumentation), may cause components, trains, or entire systems to be unavailable on demand during an initiating event. The review of the human reliability analysis (HRA) portion of the IPE examines the licensee's HRA process to determine the extent to which pre-initiator human events were considered, how potential events were identified, the effectiveness of any quantitative and/or qualitative screening processes used, and the processes used to account for plant-specific performance shaping factors (PSFs), recovery factors, and dependencies among multiple actions.

2.3.1.1 Types of Pre-Initiator Human Actions Considered

The Waterford 3 IPE considered both of the traditional types of pre-initiator human actions: failures to restore systems after test, maintenance, or surveillance activities and instrument miscalibrations. Consistent with other HRA methods, "slips" were the only pre-initiator error mode modeled.

2.3.1.2 Process for Identification and Selection of Pre-Initiator Human Actions

The submittal indicates that the "quantification as well as the identification and qualitative assessment of human failure events (HFEs) follows an SAIC technique (Dougherty and Fragola, 1988) that is nearly identical to SHARP1." In the licensees response to the NRC's request for additional information (RAI), it was stated that the pre-initiator events were included during the development of the system fault trees by reviewing the various failure modes of the systems and accounting for human induced failures. Human interactions with the equipment were examined and operating, calibration and surveillance procedures were reviewed. While discussions with plant personnel on the interpretation and implementation of procedures were not explicitly mentioned, a reasonable set of pre-initiators were listed in the submittal and it appears that relevant information sources were examined.

2.3.1.3 Screening Process for Pre-Initiator Human Actions

A screening value of 0.003 was assigned as the basic probability of a slip involving a single train of equipment, e.g., failing to restore equipment in HPSI train A. The screening probability of a slip affecting multiple trains was set at 0.0003 (e.g., miscalibration of all four SG-1 pressure sensors), which is a train or "beta factor" of 0.1. THERP was cited as the source from which these values were derived. Apparently, no pre-initiators were actually screened out. All events initially considered were included in the fault tree used for final quantification. However, events "surviving" screening were analyzed in more detail.

2.3.1.4 Quantification of Pre-Initiator Human Actions

A "time-independent" technique was used to quantify all "slips", which is assumed to be the only failure mode for pre-initiator events. The submittal states that the technique is a variant on THERP and is similar to the ASEP HRA procedure. The technique assumes a basic human failure probability (BHEP) of 0.003 and a "multiple component beta factor" to (essentially) account for common cause slips across trains. For two train systems, a multiplier (or beta factor) of 0.1 is used and for three or more trains, a beta factor of 0.01 is assigned. Thus, the common cause failure probability for three or more pressure sensors could be 3.0E-5, but is usually even less due to credit for recovery by a checker etc. While this approach provides a reasonable treatment of dependencies across trains, the use of a BHEP of 0.003 is not necessarily "conservative" as was asserted in the response to the NRC's RAI. The licensee argues that "the use of the THERP single-task value (0.003) as the probability of any slip on the train, is equivalent to assuming that all tasks involved with the train are completely dependent on the first task undertaken." This statement, however, is inaccurate. It is usually the case that a failure on any of several tasks related to the train will lead to its failure on demand. Therefore, the probability of failing on each of the critical tasks should be *added* together. If the single-task BHEP is 0.003 and there are four critical tasks, then the total failure probability would be 0.012 (4 x 0.003), at least before recovery credit is given. Thus, there was nothing particularly "conservative" about the technique used and in some cases it could be argued that the HEP values obtained for some events are optimistic. On the other hand, credit for redundant crew members (e.g., an independent post-maintenance check) was taken for only one checker and moderate dependency was assumed (a value of 0.14 was assigned in all cases). On the basis of recovery credit allowed in methods such as ASEP, the amount of credit is not unreasonable and could be somewhat pessimistic.

The response to the RAI states that no PSFs were considered in using the time-independent technique and that *restorations* across trains were assumed to be independent. While the latter assumption is reasonable

(and defended in the response to the RAI), the lack of consideration of any plant-specific PSFs, coupled with the more or less "generic" approach used to quantify the pre-initiator events, results in a HRA that may or may not provide a good representation of the actual plant. Nevertheless, at least both restoration and miscalibration events were modeled and the assigned HEPs were not unreasonable. Therefore, the pre-initiator analysis provided at least some opportunity to identify potentially important events, even if its usefulness is limited by the lack of consideration of plant -specific factors.

2.3.2 Post-Initiator Human Actions

Post-initiator human actions are those required in response to initiating events or related system failures. Although different labels are often applied, there are two important types of post-initiator human actions that are usually addressed in PRAs: response actions and recovery actions. Response actions are generally distinguished from recovery actions in that response actions are usually explicitly directed by emergency operating procedures (EOPs). Alternatively, recovery actions are usually performed in order to recover a specific system in time to prevent undesired consequences. Recovery actions may entail going beyond EOP directives and using systems in relatively unusual ways. Credit for recovery actions is normally not taken unless at least some procedural guidance is available.

The review of the human reliability analysis (HRA) portion of the IPE determines the types of post-initiator human actions considered by the licensee and evaluates the processes used to identify and select, screen, and quantify the post-initiator actions. The licensees treatment of operator action timing, dependencies among human actions, consideration of accident context, and consideration of plant-specific PSFs is also examined.

2.3.2.1 Types of Post-Initiator Human Actions Considered

The Waterford 3 IPE categorizes human actions as either human failure events (HFEs) or recovery actions. The distinction is simply "functional" in the sense that HFEs are included in the fault or event trees, while recovery actions are applied at the cutset level. HFEs included both pre- and post-initiator events, but only a few (apparently four) post-initiator events were included in the "top logic." The rest of the human actions were labeled recovery actions because they were applied to the cutsets after initial quantification. Thus, the traditional distinction between response and recovery type actions was not made in the Waterford 3 submittal.

Three criteria were identified for recovery actions: 1) the equipment to accomplish the recovery must exist and be available, 2) time to accomplish the action must be available, and 3) the action must be in procedures, taught in training, or otherwise be obvious to the operators. Given these criteria, the recovery actions could include both response and recovery actions as described in the traditional sense. However, a distinction was made between potential rule-based mistakes (procedure-based) and "response mistakes" which were described in the response to the RAI as mistakes "in on-the-spot, general diagnosis and response in the absence of rules." These two categories appear to roughly fit the traditional distinction between response and recovery actions, but in the Waterford usage, the "response" actions are the ones that may not be proceduralized. A third type, verification mistakes, was also discussed. Verification mistakes were described as mistakes in immediate actions found in the emergency procedures. These three different types of actions were treated in the analysis by assignment of different "type" factors (discussed further below). When considered independently from other factors, verification actions were more likely to succeed than rule-based, which were in turn were more likely to be successful than response actions.

In any case, at least some of the actions modeled were not proceduralized. This is defended in the licensee's response to the RAI on the grounds that the operators are trained to respond to accidents and recover critical safety functions and that credit for such actions were only taken in long-term scenarios. In many cases, the Technical Support Center would be available to assist the operators. A review of the "response" actions listed in the submittal did not suggest that extraordinary behavior was being asked of the operators, but information on the events was minimal.

2.3.2.2 Process for Identification and Selection of Post-Initiator Human Actions

The submittal and the response to the RAI indicate that all but a few of the post-initiator human actions were selected by manually reviewing cutsets and determining if operator actions could mitigate the sequence. The submittal states that "the HRA task served as an integral advisor to other project tasks to assure that relevant human interactions were identified and properly incorporated into the logic models." The HRA task was involved during initial sequence and modeling efforts and "during this period had the opportunity to review plant and system design information and become familiar with the control room and related operating procedures." While simulator exercises were not conducted, the statements discussed above suggest that the HRA analyst was significantly involved throughout the modeling effort. Thus, it appears that steps were taken to assure that appropriate human action identification and selection occurred.

2.3.2.3 Screening Process for Post-Initiator Response Actions

The response to the RAI states that screening values were used for "post-initiator top logic mistakes" and for post-initiator slips. Only two post-initiator slips were modeled and, as was done with pre-initiator slips, they were assigned a screening value of 0.003. The response to the RAI did not make clear exactly why these "slips" were modeled. The actions apparently involve operator failure to align the alternate AC power source following failure of the normal AC power source for Static Uninterruptible Power Supplies (SUPs). The licensee states that the screening value is acceptable because the realistic failure rate for these events is expected to be lower. The licensee also argues that "these SUPs recoveries are of no importance" (see page 2-14 of response to RAI), so it is not clear exactly why they were modeled, particularly as slips. In any case, all sequences containing these events were truncated and the events were left at their screening value.

As for the top logic events modeled, rule-based actions with "no burden" were assigned a screening value of 0.1. Rule-based actions "with burden" were screened at 0.2 and non-rule-based actions (response) were screened at 0.4. In the response to the RAI, the licensee argues that with a truncation value of 1.0E-9, these values are high enough to ensure that no important sequences were eliminated. The argument would be true as long as only one action was credited, but not necessarily so if multiple actions with dependencies were present. Nevertheless, since only a few top logic events were actually modeled and the cutsets were examined after initial quantification, the screening approach is probably reasonable. Moreover, apparently all the top logic events modeled were later quantified in detail.

2.3.2.4 Quantification of Post-Initiator Human Actions

The quantification of all post-initiator human actions (except the two slips discussed above) was based on the time-dependent system of time reliability correlations (TRCs) developed by SAIC and documented in the book by Dougherty and Fragola and in an American Nuclear Society conference paper by Dougherty (1989). The submittal states that the TRCs are similar to the HCR and RMIEP TRC methods.

In the response to the RAI, the basic form of the TRC is provided along with discussions regarding the relevant input parameters for both an in-control room model and an ex-control model (i.e., for actions to be performed outside the control room). Brief discussions of the input parameters were also provided in the submittal. The critical elements for the in-control room model include: the available response time and an estimate of the median response time for the event examined, along with adjustments for type of behavior (verification, rule-based, and response type, see section 2.3.2.1 above for descriptions), degree of "crew burden", success likelihood (an index that can be used to reflect the impact of PSFs), and model uncertainty. The model uncertainty factor is fixed at 1.68, apparently to reflect that the model uncertainty is distributed lognormally about the mean.

For the ex-control room model, similar parameters are modeled, along with adjustments to response time for potential "delaying hazards" outside the control room. The model uncertainty factor can also be adjusted for uncertainty due to other influences or hazards. Hazard factors which can influence response time include lighting, instrument separation, need for tools, need for protective clothing, and other miscellaneous hazards. Guidance is provided for how much time to add due to the hazards, but the basis for the selected times to be added was not provided. The model uncertainty factor can be adjusted with a multiplier for the number of hazards involved. The hazards considered for this adjustment include need for remote coordination, security access, noise, and availability of tools. The basis for the multipliers for this parameter were not provided either. In addition, two assumptions are made for the ex-control room. First, when available time is equal to mean response time, the failure probability is set to 0.5. Second, the reliability of ex-control room actions with 3 minutes response time is comparable to in-control room actions, if no other hazards (other than performance outside the control room) are present.

While it is impossible at this point to determine the overall basic validity of the method briefly described above and used in the Waterford 3 IPE (the "SAIC method"), the basic TRCs are apparently consistent with those used by other methods and the approach does attempt to provide mechanisms for addressing various factors that should influence operator performance. However, as with all HRA methods, the validity of the results can be no better than the quality of the analysis on which the analysts base their judgments. For example, to what extent were plant-specific PSFs considered and how accurate were the estimates of the timing parameters? These and other aspects related to the quality of the Waterford HRA are discussed below.

The response to the RAI indicated that all the success likelihood indices (SLIs) except two were left at their default values. That is, PSFs were assumed to have no effect on all but two events. For these two events, the SLI was increased to reflect expected improved performance. In one case for good training and in the other for many more hours being available than was assumed. By leaving the SLIs for the remaining events at their default values, the analysts are basically assuming Waterford is an "average" plant in terms of its PSFs. Other than the fact that two events were examined in enough detail to determine that the HEPs should be lower, there was no evidence that plant specific PSFs were examined for other events. The resulting analysis therefore appears to be somewhat "generic" rather than plant-specific and may or may not adequately represent the plant. At a minimum, judgments were made regarding the extent to which operators are burdened in a particular scenario and the type of task involved.

The NRCs RAI requested that examples of the application of the two calculation techniques be provided that exercised all the parameters in the techniques. On the basis of the examples provided, it would appear that many of the parameters were left at their default values. One specific example requested in the RAI was to provide a description of the application of the method to operator action ZMANTRAN,

which is the action to manually transfer (from the control room), the 6.9 and 4.16 KV buses from the UATs to the SUTs following failure of the auto fast transfer. On reason this action was selected was because of the relatively low HEP of 7.5E-6 listed for this event in the submittal. Given the values of the parameters applied, apparently the assumption of a rule-based type action in the context of 60 minutes of available time, produces relatively low HEPs in the SAIC method. No other special considerations were necessary to obtain such a value other than the operator being assumed unburdened and an assumption of "average" SLI..

In general, the way in which the SAIC HRA method was applied in the Waterford IPE did not appear to violate its basic tenets and the resulting HEPs would not in most cases be considered excessively low. The main concern in regard to the general application of the method is the extent to which plant-specific PSFs were considered. The information provided suggests that in most cases "default" values were assumed and there was no evidence that detailed analyses were performed to assure that the "generic" values were appropriate. However, the HEP values themselves would not suggest that identifcation of human action vulnerabilities was necessarily precluded. Another important factor that relates to the adequacy of the application of the method is the determination of timing parameters. This aspect is discussed next.

2.3.2.4.1 Estimates and Consideration of Operator Response Time

The determination of the time available for operators to diagnose and perform event related actions is a critical aspect of HRA methods which rely on TRCs to assess the probability of operator failure. In order to appropriately use the SAIC TRCs, the net available time for an operator to respond must be determined by considering the appearance of cues, such as control room alarms or other indications, that signal the operators that a particular response is required. In many cases the time at which operators receive the relevant cues is significantly later than when the event to be responded to actually occurred. Thus, if the point at which the relevant cues occur is not considered in determining available time, the resulting estimates could be significantly greater than the actual time available. Moreover, if significant, the time needed to perform a certain action must be subtracted from the total available time before the TRCs are used. For example, if the actions necessary to accomplish a particular task, such as the switchover to recirculation, require 15 minutes and only 30 minutes total time is available, then the operators have only 15 minutes available. Thus, 15 minutes rather than 30 minutes should be used with the TRC equation and the result is non-trivial (e.g., an order of magnitude in difference).

The submittal itself did not discuss the approach used to determine or estimate the time available for operator actions. However, the licensee's response to the NRC RAI did provide some insight. "In general, the available time was determined from applicable system response analyses. In some cases engineering judgment was used to determine the available time given the most limiting sequence." Furthermore, the response to the RAI states "that in most cases, a minimum available time is used to avoid differentiating between sequences." Obviously, such a bounding approach will produce somewhat pessimistic HEPs for some cases, but at least will not preclude identifying potentially important events.

The response to the RAI further indicates that the temporal occurrence of indicators was considered in determining available time. However, somewhat surprisingly, the licensee indicates that "no delay in the receipt of the cue to act was assumed." In response to a follow-up RAI on this issue, the licensee indicated that delays in receipt of indications in the control room were actually carefully examined, but that relative to the time available for the events of interest, the delay was insignificant and therefore not considered. While this may be appropriate for the events modeled in the Waterford IPE and clear

examples were provided by the licensee, many other IPEs have tended to find it necessary to account for delays in the occurrence of relevant cues. For actions inside the control room, the time to execute the response was also assumed negligible. Finally, a default median response time of four minutes was assumed for all of the in-control room actions modeled and adjusted according to the type of behavior involved in the task. The licensee states that the default value was derived from the nominal diagnosis curve from THERP.

Regarding ex-control room actions, the licensee states that the "human reliability" is assumed to be dominated by the actions taken outside the control room, not on the decision-making process; therefore "only the time required for the action to be carried out outside the control room is included in the ex-control room model." However, in response to a follow-up RAI, the licensee indicated that while the statement that only the time required for the actions to be carried out outside the control are included in the ex-control room model was true with regard to the user inputs to the model, the model itself, and the resultant HRA failure probabilities are calibrated to a TRC presented in NUREG/CR-2787 (Interim Reliability Evaluation program (IREP) - Analysis of Arkansas Nuclear One, Unit One Nuclear Power Plant), which includes *all* required actions to perform the recovery. Thus, the licensee argues that the TRCs used to determine the HEPs do take the diagnosis and decision time into account.

A list of the response times assumed for the ex-control room actions and additions to response times based on delay hazards was presented in the licensee's response to the RAI. The response times were determined "from interviews with operators" and "the presence of hazards which could influence the response time and uncertainty was natural outcome of these operator interviews." Exactly how many operators were interviewed and the approach for soliciting the estimates were not discussed. Other methods, such as THERP have argued that time estimates obtained from operators should be doubled, but this is not mentioned by the licensee. Without additional detail, it is difficult to determine whether or not the response times used are reasonable. Regardless, the total time assumed available tends to be substantially longer than the estimated response time and the HEPs do not in general appear to be excessively low.

2.3.2.4.2 Other Performance Shaping Factors Considered

Other than those discussed above, there was no evidence of any other PSFs being considered.

2.3.2.4.3 Consideration of Dependencies

Two basic types of dependencies are normally considered in quantifying post-initiator human actions: 1) time dependence and 2) dependencies between multiple actions in a sequence or cut set. One type of time dependence is concerned with the fact that the time needed to perform an action influences the time available to recognize that a problem has occurred and to diagnose the need for an action. This type of time dependence is handled by the Dougherty and Fragola method by using TRCs which reflect the likelihood of operators diagnosing *and* performing the related actions in a particular time window. In essence, the method assumes that the probability of errors in performing in-control room actions is negligible compared to the potential for diagnosis failure. Moreover, the response times for ex-control room actions are assumed to be dominated by the actions taken outside the control room, not on the decision-making process. The validity of this assumption is certainly debatable.

Another aspect of time dependence is that when sequential actions are considered, the time to complete one action will impact the time available to complete another. Similarly, the sooner one action is

performed, the slower or quicker the condition of the plant changes. This type of time dependence is normally addressed by making conservative assumptions with respect to accident sequence definitions. One aspect of this approach is to let the timing of the first action in a sequence initially minimize the time window for subsequent actions. The occurrence of cues for later actions are then used as new time origins. This type of dependence was apparently handled in the same way as other context effects and is discussed below.

The second type of dependence considers the extent to which the failure probabilities of multiple human actions within a sequence or cutset are related. There are clearly cases where the context of the accident and the pattern of successes and failure can influence the probability of human error. Thus, in many cases it would clearly be inappropriate to assume that multiple human actions in a sequence or cut set would be independent. Furthermore, context effects should be examined even for single actions in a cut set. While the same basic action can be asked in a number of different sequences, different contexts can obviously lead to different likelihoods of success. Dependence among multiple human actions was handled in the Waterford submittal essentially by assuming that they are independent. The licensee argues that "between separate tasks independence is provided because many of the tasks are performed by different people, and there is separation in time or "cognitive space", i.e., cues are independent enough to force subsequent diagnosis." The licensee further states "that context effects were handled by lumping the different sequences into one event." "This is done by using a sum average time for the available time parameter for events that are sequence dependent."

2.3.2.4.4 Quantification of Recovery Type Actions

The submittal indicated that all post-initiator human actions were quantified with the approach described above in section 2.3.2.4. Different TRC parameters were used to quantify non-rule-based as opposed to rule-based actions.

2.3.2.4.5 Human Actions in the Flooding Analysis

In the Waterford 3 IPE, human actions and human recovery of several flooding scenarios were modeled. During initial quantification (screening) all ex-control actions were set to fail. In addition, in-control room actions for those flood scenarios that started or propagated through the control room were also assumed to be failed as considered. All the actions modeled initially were identical to those modeled in the Level 1 analysis and the flooding analysis "caused no special requantification of level 1 human actions." After the initial screening, consideration was given as to whether any human recovery actions which were set to 1.0 could be assumed to be performed under conditions of the flood. Any human actions (inside or outside the control room) with some dependency on flood or flood disabled equipment were simply assumed to fail. Other wise, the Level 1 HEPs were used. Apparently during the early rounds of quantification, a flood recovery value of 0.01 was applied to large flood scenarios. Later, three new recoveries were created for the flooding analysis and were quantified using the EPRI draft report on "Modeling of Recovery Actions in PRAs." The actions included: 1) isolating the flood before ex-control room actions or equipment are disabled, 2) a local action to recover an in-control failure (or inability) to align the CST to the CSP as an inventory makeup source, and 3) a local action to recover an in-control failure (or inability) to stop the RCPs within 30 minutes of the loss of seal cooling.

The quantification of these actions was documented in the response to the RAI and appeared reasonable. Per the EPRI method, time available, training, task complexity, and environmental factors were all

considered. The treatment of human actions in the flooding analysis was relatively thorough and reasonable.

2.3.2.4.6 Human Actions in the Level 2 Analysis

The licensee states that human actions were not credited in the Waterford 3 Level 2 analysis.

2.3.2.5 Important Human Actions

The Waterford 3 submittal presents a list of basic event importance as determined by Fussell-Vesely (F-V) measures. Operator actions with F-V values greater than 0.01 (1% of CDF) are presented in Table 9 below, along with their F-V values and their HEPs. The sensitivity analysis performed by the licensee also examined which cut sets fell below the reporting criteria due to human recoveries. The operator actions identified included the initiation of cooldown for a SGTR, the stopping of the RCPs within 30 minutes of a loss of seal cooling (loss of instrument air initiator), and the closing of miniflow valves during recirculation mode for medium and large LOCAs and for the loss of instrument air initiator.

Table 9 Important Human Actions

Event Description	F-V	HEP
Operator fails to recover from room cooling failure (ZHFHVACREC)	8.01E-02	5.0E-01
Operator failure to align HPSI pump train AB (ZHFOPALNAB)	4.68E-02	9.8E-02
Operator fails to align EFW suction to WCT (Recov. Action) (ZEFWWCT)	3.38E-02	1.6E-01
Operator fails to align EFW suction to WCTs following LO LVL (ZEFWWCT-1)	2.43E-02	1.5E-02
Operator fails to restore air cooling unit after test/maint (UHF25AREST)	1.57E-02	3.0E-03
Operator fails to restore air cooling unit after test/maint (UHF30AREST)	1.57E-02	3.0E-03
Operator fails to restore air cooling unit after test/maint (UHF30BREST)	1.28E-02	3.0E-03
Operator fails to restore air cooling unit after test/maint (UHF25BREST)	1.28E-02	3.0E-03
Operator fails to manually initiate RAS small/large LOCA (ZMANRASA-S)	4.98E-03	1.5E-01

2.4 Back End Technical Review

2.4.1 Containment Analysis/Characterization

2.4.1.1 Front-End Back-End Dependencies

The interfaces between the front-end and back-end analyses are provided in the IPE by the definition of 10 plant damage states (PDSs). An event tree structure (called Level 1 to Level 2 bridge tree in the IPE submittal) is used in the Waterford 3 IPE to sort out Level 1 core damage sequences and combine them with containment system status for PDS definition (Section 4.3 of the IPE submittal). The parameters used in the IPE to define the PDSs include:

- AC Power Availability,
- Containment Integrity Status,
- RCS Pressure,
- Core Melt Timing,
- Containment Mitigating Systems.

Except for SBO and bypass PDSs, the PDSs defined in the Waterford IPE are based on RCS pressure, which depends on the type of accident sequences (or accident initiators), the time of core melt, which depends on whether core cooling is lost during the injection or the recirculation phase, and the availability of containment systems. The conditional probabilities for the PDSs at various RCS pressures (or types of accidents) are: 39% for PDSs with medium RCS pressure (from small or medium break size LOCAs or transient initiated events with stuck-open pressurizer safety relief valves); 15% for PDSs with high pressure PDS (from total loss of all feedwater transient); and 1% for PDSs with low pressure PDS (from large LOCAs). In addition to the above PDSs, the conditional probability for SBO sequences is 38%, the conditional probability for SGTR sequences is 5%, and the conditional probability for ISLOCA sequences is 3%.

For individual PDSs, the most probable PDS is PDS IH (21% CDF), a PDS with medium RCS pressure, early core melt, and failure of containment heat removal. This is followed by a SBO PDS with early core melt (21%), a SBO with late core melt (17%), and PDS IIIB (15% CDF), a transient with early core melt, but with containment heat removal.

The PDSs defined in the Waterford 3 IPE are of sufficient detail to provide a proper account of the front-end and back-end dependencies and adequate information for back-end accident progression analysis.

2.4.1.2 Containment Event Tree Development

Probability quantification of severe accident progression is performed in the IPE by the use of containment event trees (CETs). The development of the CETs is discussed in Sections 4.5 of the IPE submittal. Four different CETs are developed for (1) transients and LOCAs (a "normal" CET), (2) SBO, (3) SGTR, and (4) ISLOCA. The CETs includes the following top events:

1. Plant damage state,
2. RCS depressurized before vessel breach,
3. Coolant recovered in-vessel before breach ,

4. In-vessel steam explosion,
5. No vessel failure,
6. No early containment failure,
7. Coolable debris formed ex-vessel,
8. AC power recovered late
9. No late containment failure,
10. Fission product removal.

Figures 4.6-1 through 4.6-4 of the submittal show the structures of the four CETs. In general, the CETs developed in the Waterford 3 IPE are well structured and easy to understand. The top events of the CET cover the important issues that determine the RCS integrity, containment response, and eventual release from the containment.

Fault trees (called logic trees in the IPE submittal) are used in the IPE to quantify the top events of the CETs. The logic trees used for CET quantification are very detailed and address all phenomena and systems important for Level 2 accident progression. The quantification of the basic events in the logic trees is based on the review of the industry literature and plant-specific analyses using MAAP code. According to the IPE submittal, the values used in the quantification are "relative values" meant to provide insights on containment performance during a sever accident. The basic events are assigned probability values based on the likelihood of occurrence. For example, a basic event is assigned a probability value of 0.8 if it is judged likely to occur. In general, the quantification process used in the IPE is systematic and traceable. Although the values assigned in the IPE seem adequate, their adequacy cannot be verified in this technical evaluation report because of the limited scope of this evaluation. Some items that are of interest are discussed in the following.

In-Vessel Recovery

The Waterford 3 IPE considers in-vessel recovery due to the injection of low pressure systems after RCS depressurization. The mechanisms for RCS depressurization considered in the IPE include that from hot leg or surge line creep rupture. Hot leg temperature calculated by MAAP, as well as information obtained from NUREG-1150, are used in the IPE to determine the probability of hot leg failure. In additional to the above in-vessel recovery mechanism, the logic trees for in-vessel recovery also include basic events for RCS depressurization and recovery of injection systems by operator actions. However, credit is not taken for these events in CET quantification.

Besides the recovery of low pressure injection, prevention of vessel breach by ex-vessel cooling is also considered in the IPE. According to the IPE submittal, the reactor cavity will be filled with water prior to vessel failure in almost all cases. This will submerge the reactor lower head and may prevent vessel failure. The probability of successful ex-vessel cooling (such that vessel breach is avoided) is assigned values of 0.75 to 0.9 in the IPE. A sensitivity study with the probability values changed to 0.20 shows that while the probabilities for both early and late failures increase with the decrease of in-vessel recovery by ex-vessel cooling (due to the higher probability of vessel failure), the effect is not significant (Response to RAI Level 2 Question 2).

Early Containment Failure

Early containment failure is defined in the Waterford IPE as that occurs at or shortly after vessel breach time. The failure mechanisms addressed in the CET logic trees for early containment failure, include:

- In-vessel steam explosion (alpha mode failure),
- Ex-vessel steam explosion,
- Early leak due to small isolation failure or missiles,
- Early rupture due to large isolation failure,
- Early rupture due to reactor vessel blowdown (rocket),
- Steam overpressure before core melt (no heat removal),
- High pressure melt ejection (HPME) effects, such as DCH,
- Reactor cavity wall failure, and
- Combustion of hydrogen prior to or during vessel breach.

The above list includes all the important early containment failure modes discussed in NUREG-1335. Quantification of containment failure for the above failure modes is based on data available in the literature, primarily those associated with NUREG-1150 analyses, and plant-specific results from MAAP code calculations. Results of sensitivity study on some of the mechanisms that involve significant uncertainties (e.g., hydrogen burn and DCH) are reported in the IPE submittal and the licensee's response to RAI Level 2 Questions.

There are a few Waterford 3 plant-specific features that may affect the probability of early containment failure. A key feature of the Waterford 3 reactor vessel is that all core instrumentation is routed from the top of the vessel and there is no instrumentation tunnel to provide access from the reactor cavity to the upper containment volume. The reactor cavity of Waterford 3 is open to the upper containment through the very small annulus between the vessel and the cavity wall and to the steam generator compartments through the RCS pipe penetrations through the cavity wall. Besides these areas, the reactor cavity also communicates with the containment volumes through a relatively small tunnel which connects to the ductwork that provides reactor cooling. This area, according to the IPE, allows the water collected in the containment sump to flood the reactor cavity. In addition to the above plant-specific features, the use of a steel shell containment, which is vulnerable to direct attack by the hot core debris, and the greater amount of Zircalloy in the Waterford 3 fuel assemblies (than in the NUREG-1150 plants), which result in the production of more hydrogen, are other plant-specific features that may affect the probability of containment failure.

Of the plant-specific features discussed above, the lack of bottom head penetrations makes a circumferential failure of the vessel bottom head more likely, and as a result, the challenge to containment integrity due to high pressure melt ejection may be more severe. On the other hand, the tight reactor cavity of Waterford 3 tends to limit the amount of debris expelled to the containment air space. For other early challenges, the amount of hydrogen in the Waterford 3 containment during a severe accident is greater than that in the NUREG-1150 containments because of the greater amount of Zircalloy in the Waterford 3 core assemblies and the longer time to vessel failure for Waterford 3 due to the lack of instrument penetrations in the lower head. The effects of these special features on containment failure are addressed in the IPE.

Containment Bypass and Induced Steam Generator Tube Rupture (ISGTR)

Temperature induced SGTR (ISGTR) is considered in the IPE both as a containment failure mechanism and a RCS depressurization mechanism. However, ISGTR as a containment failure mechanism is not discussed in the IPE submittal, it is discussed in the licensee's response to RAI (Level 2 Question 6). According to the response, a separate SGTR PDS, not reported in the IPE submittal, is used for ISGTR. The ISGTR PDS is created in the IPE by performing a logical AND operation of the ISGTR probability

with the Level 1 transient sequences that involve high RCS pressure and dry secondary side. Since it is considered in the IPE that it is much more likely that the hot leg will rupture before the steam generator tubes, a relatively high probability of the tubes remaining intact is used in the IPE. The Waterford 3 IPE also does not differentiate the induced tube rupture probability between RCPs running or not running. It is argued in the response to the RAI that, with the RCPs running, the hot leg would also heat up faster so that the relative probability of hot leg rupture versus SGTR is expected to remain about the same.

Debris Coolability and Late Containment Failure

The failure mechanisms addressed in the CET logic trees for late containment failure include:

Late rupture due to:

- Hydrogen combustion, and
- Reactor cavity wall failure (caused by CCI),

Late Leak due to:

- Steam generation,
- Non-condensable gas generation,
- High temperature failure of elastomer penetration seals, and
- Basemat melt-through.

The above list includes all the important late containment failure modes discussed in NUREG-1335. Similar to early containment failure, quantification of containment failure for the above failure modes is based on data available in the literature, primarily those associated with NUREG-1150 analyses, and plant-specific analysis results using MAAP code. Results of a sensitivity study on debris coolability are reported in the licensee's response to RAI Level 2 Question 11.

In the Waterford 3 IPE, three basic events are used to address the probabilities of debris coolability under the following conditions: HPME, ex-vessel steam explosion (EVSE), and no-HPME and no-EVSE. The probabilities of debris coolability used in the Waterford 3 IPE for the above three conditions are 0.6, 0.5, and 0.8, respectively. It is therefore assumed in the Waterford 3 IPE that the debris is more likely to be in a coolable condition if HPME and EVSE do not occur. This order is not consistent with that obtained in the NUREG-1150 study. In NUREG-1150, the core debris in the reactor cavity is more likely to be coolable for HPME and EVSE (0.8 coolable probability) than for cases with no HPME and no EVSE (0.35).

According to the Waterford IPE, the probability of debris coolability is lower for HPME and EVSE because fragmentation of the debris by HPME and EVSE causes the debris to break up into small particles, which can group tighter together to form a solid mass that water cannot penetrate as easily as for large particles. In NUREG-1150, in addition to particle size, the effect of spreading the debris outside of the reactor cavity is also considered. In is noted that, in comparison, debris dispersing is more restricted in Waterford 3 than in the NUREG-1150 plants because of the lack of an instrument tunnel and a tighter reactor cavity for Waterford 3.

Core concrete interaction (CCI) occurs if ex-vessel debris is not coolable. The containment failure mechanisms considered in the IPE for CCI include those associated with non-condensable gas generation

and basemat melt-through. A containment failure probability of 0.005 is assigned to both of these mechanisms in the Waterford 3 IPE. According to the licensee's response to RAI (Level 2 Question 11), the amount of the noncondensable gases generated by CCI is not sufficient to challenge containment integrity even if the basemat is penetrated, and basemat melt-through is also not likely to occur because of the thickness of the basemat (about 10 feet above containment liner) and the low penetration depth predicted by MAAP within a 24 hour mission time.

The most important late containment failure mode in the Waterford IPE is that by steam generation. According to the IPE, containment fails by steam pressurization within the mission time if both containment spray and containment fan coolers fail. Furthermore, the water collected in the reactor cavity from lost RCS inventory is sufficient to pressurize the containment to failure pressure.

Fission Product Removal

Credit is taken in the IPE for all fission product removal mechanisms considered in NUREG-1150 (i.e., those incorporated in the XSOR code).

2.4.1.3 Containment Failure Modes and Timing

The Waterford 3 containment ultimate strength evaluation is described in Section 4.4.2 of the IPE submittal. The ultimate containment failure pressure for the Waterford 3 IPE is estimated by hand calculation of stress at 1% strain level and comparison to existing analyses of structural capacity for similar plants. The ultimate pressure obtained for the Waterford 3 containment is 135 psig. The containment failure pressure distribution used in the Waterford 3 IPE is a log-normal distribution with a medium failure pressure of 135 psig and a coefficient of variation of 0.15.

In comparison with the distributions obtained and used in NUREG-1150, the pressure distribution used in Waterford 3 IPE is much flatter. The distribution provided in Figure 4.4-1 of the IPE submittal for failure probability versus pressure (i.e., the fragility curve) is almost linear from 40 psig (approximately the design pressure) to 135 psig (the mean failure pressure). The use of this distribution seems to contribute to the relatively high containment failure probability for early containment pressure loads predicted in the IPE. For example, according to Table 4.6-3 of the submittal, the containment failure probability is 0.286 for a containment pressure load of 89 psia. According to the licensee's response to the RAI (Level 2 Question 7), a change to the probability curve was made subsequent to the submittal of the IPE, and the new curve is more consistent with that used in NUREG-1150 and other IPEs. According to the new curve, the containment failure probability for a 89 psia pressure load is 0.005. The licensee's response to the RAI also presents the containment failure results (in terms of containment release category, CRC) obtained from the use of the revised distribution curve. (The containment failure distribution provided in the licensee's response to RAI Question 7 is incorrect. The correct distribution is provided in the licensee's response to a follow-up RAI.)

The containment failure pressure obtained in the Waterford 3 IPE and the revised distribution curve reported in the RAI response seem to be consistent with those obtained in other IPEs. The original distribution curve presented in the IPE submittal seems to be overly pessimistic in predicting containment overpressure failure probabilities.

2.4.1.4 Containment Isolation Failure

Containment isolation failure is evaluated in the Waterford 3 CET under top event CFE (Containment fails early) and is addressed in the associated logic trees. It is stated in the submittal that "The probability is determined by solving a separate fault tree for failure to isolate these penetrations." (p4.6-7) However, the fault trees for isolation failure are not provided and details are not discussed in the IPE submittal.

According to the IPE submittal, both small and large isolation failure are considered in the IPE. The main small isolation problem is the failure to close small valves, e.g., the primary sampling system containment isolation valves. Overall, the probability of small isolation failure is less than 2E-5 per year (1.88E-5 in the IPE). Large isolation failures include containment penetrations that are 2 inches in diameter or larger. The dominant large isolation problem is mechanical failure of the penetration itself. A conservative high screening value of 1E-3 per year is used in the IPE.

Since details on containment isolation failure are not presented in the IPE submittal, a question is asked in the RAI (Level 2 question 10). In the response to this RAI question, the licensee discusses the analysis of containment isolation failure performed in the Waterford 3 IPE in terms of the five areas identified in the Generic Letter. According to the descriptions provided in the IPE submittal and the licensee's response to the RAI, all five areas identified in the Generic Letter regarding the evaluation of containment isolation failure are addressed in the IPE.

2.4.1.5 System/Human Responses

Although the logic trees include basic events for RCS depressurization and recovery of injection systems by operator actions, credit is not taken for these events in the CET quantification. On the other hand, AC power recovery is the primary reason that only 50% of all SBO sequences result in containment failure. In the IPE, the values used for power recovery are based on the same AC power recovery curve used in the Level 1 analysis. For the Level 2 analysis, two additional decisions are included in the SBO CET for (1) AC power recovery before vessel breach, and (2) AC power recovery before containment failure The treatment of the additional time for AC power recovery in the IPE seems reasonable.

2.4.1.6 Radionuclide Release Characterization

The end states of the CET (defined as containment release categories, or CRCs, in the Waterford 3 IPE submittal) are discussed in Section 4.6.4 of the IPE submittal. The following issues are used to define a CRC:

1. Did the reactor vessel remains intact?
2. Did HPME occur at vessel breach?
3. Did the containment fail at all?
4. Did the containment fail any time before, or soon after vessel breach?
5. Did the containment fail long after vessel breach?
6. Was the containment failure sudden or gradual, i.e., a leak or a rupture?
7. Was any ex-vessel debris cooled?
8. Did sprays wash fission products out of the containment atmosphere?

These cover the vessel failure status, the containment failure mode, CCI, and fission products scrubbing by containment sprays. A total of 76 CRCs are defined in the IPE (Table 4.7-2 of the submittal) and

source terms are defined for 60 CRCs (Table 4.7-3). CET results show 11 CRCs with non-zero frequencies (Table 4.8-1). From the description provided in the IPE submittal it seems that the CET end state grouping for source term definition in the Waterford 3 IPE is adequate.

The CET quantification results provided in Table 4.8-1 of the IPE submittal show 11 CRCs. Among the 11 CRCs are one bypass CRC, which can be further divided to one CRC with SGTR and another with ISLOCA, 4 early failure CRCs, 5 late failure CRCs, and one no failure CRC. The percentage contributions of these CRCs to the total CDF are 46% for no failure, 20% for late failure, 26% for early failure, and 8% for bypass failure. For bypass failure, the conditional probability of SGTR is about 5%, primarily from SGTR as an initiating event, and the conditional probability of ISLOCA is 3%.

Source terms for the CET end states are determined by accident progression analyses using a method similar to that used in NUREG-1150 studies. Source terms obtained in the IPE are presented in Tables 4.7-3 and 4.8-3 of the IPE submittal. Source terms are presented in these tables in terms of release fractions noble gases, Iodine, Cesium, Tellurium, and Strontium.

The release fractions predicted in the IPE for the SGTR sequences (CRC BP-E5A) are much less than those for some early failure sequences. This is because of the assumed availability of water scrubbing for the SGTR sequences in source term calculation. Since water scrubbing may not be available for all SGTR sequences, the release fractions reported in the submittal for SGTR sequences may not be adequate for some SGTR sequences. According to the licensee's response to RAI (Level 2 Question 16) there are SGTR sequences where water scrubbing is not available, and contribution from these SGTR sequences is not significant. However, quantitative data on the relative contributions from the different SGTR sequences are not provided in the response. Since the release fractions for the SGTR sequences without water scrubbing are expected to be much greater than those with water scrubbing, the omission of the source term for SGTR without water scrubbing is very optimistic. Although it is not a significant problem in the present IPE because of their small frequencies in comparison with those of other sequences that have large releases (e.g., ISLOCA), it is a deficiency nonetheless. It would be desirable to divide the SGTR CRC to two CRCs with and without water scrubbing and to obtain the source terms for both of them. This would assure that significant information is not lost in the IPE process in the future IPE update.

2.4.2 Accident Progression and Containment Performance Analysis

2.4.2.1 Severe Accident Progression

In the Waterford 3 IPE, the MAAP code was used to develop information to assign basic event and containment failure probabilities. The sequences that are calculated by the MAAP include those associated with (1) large break LOCA, (2) small break LOCA, (3) total loss of feed water, and (4) containment bypass. In general, the sequences selected for MAAP calculations are the dominant Level 1 sequences in the PDSs. According to the licensee's response to RAI (Level 2 Question 3), there is not much difference in the Level 1 sequences within a PDS because of the way the Waterford 3 PDSs are constructed.

The sequences selected for source term analyses and the source terms definition used in the IPE seem to be adequate.

2.4.2.2 Dominant Contributors: Consistency with IPE Insights

Containment release categories (or containment failure modes) and their frequencies obtained from the Waterford 3 CET quantification are discussed in Section 4.8 of the submittal. Table 10, below, shows a comparison of the conditional probabilities for the various containment failure modes obtained from the Waterford 3 IPE with those obtained from the Surry and Zion NUREG-1150 analyses.

Two sets of data are presented in Table 10 for Waterford 3: one from the IPE submittal based on the us of a very conservative containment fragility curve, and the other from the licensee's response to the RAI using a revised containment fragility curve more consistent with those used in other IPEs.

Table 10 Containment Failure as a Percentage of Total CDF

Containment Failure Mode	Waterford 3 IPE+	Waterford 3 IPE Udpate++	Surry NUREG-1150	Zion NUREG-1150
Early Failure	26	4	0.7	1.4
Late Failure	20	25	5.9	24.0
Bypass	8	8	12.2	0.7
Isolation Failure	***	***	*	**
Intact	46	63	81.2	73.0
CDF (1/ry)	1.7E-5	1.7E-5	4.0E-5	3.4E-4

\+ The data presented for Waterford 3 are based on Table 4.8-1 of the IPE submittal.
++ Data presented in this column are those obtained from using a revised containment fragility curve (reported in the response to a follow-up RAI).
* Included in Early Failure, approximately 0.02%.
** Included in Early Failure, approximately 0.5%.
*** Included in Early Failure, approximately 0.1%.

As shown in the above table, the conditional probability of containment bypass for Waterford 3 is 8% of total CDF. Of the 8% bypass probability, 5% comes from steam generator tube rupture and 3% comes from ISLOCA. The contribution from ISGTR is small and not reported separately in the IPE submittal.

The conditional probability of early containment failure presented in the IPE submittal is about 26% (of total CDF). the major threat to early containment failure is a loss of containment heat removal during an accident where the RCS is at high pressure. Since the containment is at elevated pressure due to steam generation a high pressure melt ejection (HPME) can challenge containment integrity. This scenario occurs during SBO and small LOCA with loss of both safety injection and containment heat removal (CHR). Of the 26% early failure probability, over 13% is from SBO sequences and over 11% is from small LOCA sequences. On a conditional basis, about 35% of SBO sequences result in early failure and about 30% of small LOCA sequences result in early failure. According to the licensee's response to RAI follow-up questions, although the probability of early containment failure is significantly reduced by the

use of a revised containment fragility curve (from 26% to 4%), the dominant sequences that lead to early containment failure remain the same as that described in the IPE submittal.

The conditional probability of late containment failure presented in the IPE submittal is 20%. The major contributor to late containment failure is steam overpressurization when CHR is lost. SBO does not contribute as much to late containment failures because of the high likelihood of AC recovery (before containment failure). Of the 20% late failure probability about 15% is from small LOCA, 4% from SBO, and 1.3% from other transients. On a conditional basis, about 39% of small LOCA sequences, 12% of large LOCA sequences, 10% of small LOCA sequences, and 9% of other transients result in late failure. According to the licensee's response to the RAI, the conditional probability of late containment failure increased from 20% to 25% when the revised containment fragility is used. Since detailed data are not provided in the RAI responses, contributions from the various accident sequences to late containment failure cannot be obtained. It seems that the increase in late containment failure probability is primarily due to the decrease of early containment failure probability, and the dominant sequences that lead to late containment failure remain the same as that described in the IPE submittal.

2.4.2.3 Characterization of Containment Performance

As shown in Table 2, for Waterford 3 Steam Electric Station, the core damage frequency (CDF) is lower than that obtained in NUREG-1150 for Zion and Surry. Except for early containment failure, the conditional probability of other containment failure modes are consistent with those obtained in NUREG-1150 for Surry and Zion. The high early failure probability can be partially attributed to the more pessimistic containment failure probability distribution used in the Waterford 3 IPE.

The C-Matrix, which shows the conditional probabilities of CET end states (or containment failure modes) for the plant damage states (or PDSs), can be obtained from the data presented in Table 4.8-1.

2.4.2.4 Impact on Equipment Behavior

The effects of harsh environment conditions on the operation of containment fan coolers are addressed in the IPE by a few basic events in the CET logic trees. The conditions that are considered in the IPE for the operation of fan coolers include those due to hydrogen burns, HPME, and post core uncovery environment. The effect of environmental conditions on containment spray is also considered in the IPE, but its effect is considered only for the determination of fission product removal and not for late containment failure. According to the licensee's response to RAI (Level 2 Question 12), the failure of containment spray due to harsh environmental conditions is considered credible only very late in an accident in the fission product release phase. It is not considered for debris cooling and containment failure because the harsh environment is not expected to affect the CS pipe until the stress forces have worked on the pipe for a long time.

2.4.2.5 Uncertainties and Sensitivity Analysis

Sensitivity studies are discussed in Section 4.9 of the IPE submittal. The sensitivity studies provided in the IPE submittal address the uncertainties associated with the following phenomena:

- High temperature rupture of the hot leg during medium pressure scenarios,
- Ex-vessel cooling,
- Ultimate containment pressure,

- Reactor cavity wall structure failure during HPME, and
- Frequency of dominant PDSs IH and SBO.

In addition to the above sensitivity analyses, Waterford 3 also performed some sensitivity analyses with the MAAP code to ensure that a broad spectrum of possible outcomes were covered (p4.2-3). The issues that were investigated by MAAP analyses include (1) in-vessel hydrogen production, (2) direct containment heating, (3) debris bed coolability, and (4) vessel failure penetration radius. General results of these sensitivity analyses are discussed in Section 4.2.3 of the IPE submittal. Results from the sensitivity cases are presented in the submittal to show the uncertainty of individual issues on some containment parameters (e.g., the uncertainty of DCH on containment pressure load). Recognizing the uncertainty in various severe accident phenomena and how the accident progression can be affected, Waterford 3 performed some sensitivity analyses with the MAAP code to ensure that a broad spectrum of possible outcomes were covered (p4.2-3). The issues that were investigated by MAAP analyses include (1) in-vessel hydrogen production, (2) direct containment heating, (3) debris bed coolability, and (4) vessel failure penetration radius. General results of these sensitivity analyses are discussed in Section 4.2.3 of the IPE submittal. Results from the sensitivity cases are presented in the submittal to show the uncertainty of individual issues on some containment parameters (e.g., the uncertainty of DCH on containment pressure load). However, their effects on containment release profiles are not discussed in the IPE submittal but are addressed in the licensee's responses to RAI questions. Additional sensitivity analyses reported in the licensee's response to RAI questions include those associated with the challenges to containment integrity by hydrogen combustion, DCH, ex-vessel debris coolability, and hot leg creep rupture for high pressure scenarios.

The sensitivity studies provided in the Waterford 3 IPE seem to have addressed the issues of significant uncertainties in the IPE analysis.

2.5 Evaluation of Decay Heat Removal and Other Safety Issues

2.5.1 Evaluation of Decay Heat Removal

2.5.1.1 Examination of DHR

The IPE addresses decay heat removal (DHR). DHR is defined as those systems required for primary and secondary inventory control and heat transfer from the RCS to an UHS following shutdown of the reactor for transients and small LOCAs. Several methods of DHR are mentioned, including the main feedwater system, the auxiliary feedwater system, the EFW system, the condensate system (in conjunction with secondary depressurization using the turbine bypass or the atmospheric dump system) and HPSI, for small LOCA inventory control.

DHR function loss contributes 1.4.E-5/yr to the CDF and is thus below the 3.0E-5/yr criterion used to define acceptably low DHR failure frequencies in NUREG-1289.

Contribution to DHR-loss CDF from the DHR frontline systems and their support systems is calculated and presented in RAI responses. Contribution of components and support systems to each DHR system's unavailability is not calculated or readily available. The DHR system contribution to DHR loss CDF is as follows (not including support system failure): EFW (40.2%), HPSI (19.8%), MFW (0.6%), main steam (0.1%) and charging (0.1%). The support system contribution is as follows: AC power (68.1%),

ACCW (15.1%), CCW(13.9%), HVAC (8.2%), DC power (2.2%), ESFAS (2.2%) and instrument air (0.2%). These percentages from RAI responses are somewhat at odds with Figure 1 as far as absolute numbers are concerned.

2.5.1.2 Diverse Means of DHR

The IPE evaluated the diverse means for DHR, including: MFW, AFW, EFW, condensate, steam relief, HPSI and charging. Cooling for the RCP seals was taken into account. In addition, containment cooling was addressed.

2.5.1.3 Unique Features of DHR

The unique features of Waterford 3 that pertain to the DHR function are as follows:

- There is no feed and bleed capability at this plant. No pressurizer PORV exists and the HPSI/charging pumps do not have the requisite head to lift the safety valves.

- The turbine driven main feedwater pumps will continue to run for most transients, as the pump flow output is automatically matched to the decay heat level.

- There are two motor driven (capacity 350 gpm each) and one turbine driven (capacity 700 gpm) EFW pump. The EFW system is automatically started and controlled. In addition, a manually started AFW pump is also available, should the other three pumps fail (the AFW pump is normally used during startup/shutdown operations). According to the submittal and the RAI responses, the turbine driven EFW pump can be expected to continue to operate with low quality steam or even water at the turbine inlet. However, this is not credited in the analysis, and the TDEFW pump is assumed failed at the time of battery depletion.

- The normal EFW suction source is the inventory in the condensate storage pool (CSP), good for about 10 hours. A backup supply are the two wet cooling tower basins, each holding about the same amount of water as the CSP. A third option is the non-seismically qualified condensate storage tank (CST) and its transfer pump.

- The EFW control valves fail open on loss of instrument air, and there is also a backup nitrogen accumulator supply in case of loss of instrument air. The turbine driven EFW pump does not require room cooling (according to calculations, RAI responses), whereas the motor driven EFW pumps do.

- Apparently the TDEFW pump can operate with low quality steam or even water at the turbine inlet. This is not credited in the analysis.

- The DC battery (battery AB) supplying control to the TDEFW pump has a SBO depletion time of 4 hours with proceduralized load shedding (1 hour without load shedding), according to the submittal. Since the IPE, the safety related batteries have been replaced with higher capacity batteries (to allow for aging), and a new non-safety battery has been installed to take up the non-safety loads serviced by the AB battery. These modifications have extended the AB battery depletion time to 6 hours.

- Condensate pumps may be used to provide feedwater to the steam generators, provided the secondary system has been depressurized to 500 psia. There are three parallel condensate pumps. The condenser hotwells have enough inventory to supply the condensate pumps for 24 hours.

- There are multiple pathways for secondary steam relief: 6 turbine bypass valves, 2 atmospheric dump valves and 6 safety relief valves.

- The RCP seals are the Byron Jackson type, which according to the submittal can sustain loss of CCW for 30 minutes (verified by tests), without tripping the RCPs; the operators are instructed to trip the RCPs immediately upon loss of CCW. CCW cooling is the only type of cooling for these seals (no seal injection provided). Because of the 4 stage seal design, and the new resistant material for seal faces, no spurious seal failures (i.e., initiating event seal LOCA) are assumed possible with these seals (consequential failures are allowed).

- There are three trains of HPSI, CCW, AC safety buses and DC safety buses. The AB buses and AB trains are functionally related, e.g., the AB train of CCW cools the AB train of HPSI, and both are supplied AC power from the AB safety bus. The third HPSI pump must be manually started on SI.

- There are also three trains of HVAC chillers. The charging pumps also have three trains (these are considered in the PRA analysis to feed the auxiliary pressurizer spray, for emergency boration in ATWS and for RCS inventory control in an SGTR). Other safety equipment has two trains. The two trains of the instrument air compressors are backed up by the three trains of the station air compressors (see below).

- There are two EDGs. The EDGs need cooling by CCW, ventilation by dedicated fans and DC power provided by the station batteries. A diesel compressor has been added to the plant post-IPE, to help in case of problems with startup compressed air.

- There are three plant batteries, A, B, and AB. The AB battery is used for TDEFW pump control in SBO conditions. As stated above, the capacity of this battery has been increased and a non-safety battery added to pick up non-safety AB loads, such that SBO depletion time of this battery is now 6 hours. The A and B batteries have also been similarly affected, such that their SBO depletion time is now 4 hours even without load shedding. Each battery is supported by two chargers.

- There is no service water system at this plant. Instead, the ultimate heat sink is provided by the dry cooling towers. As there are multiple fans in the towers, they can be maintained piecemeal, such that maintenance would not disable the whole tower (although in the IPE it is conservatively assumed that it does). Also, in case of increased demand (depending on air temperature) and during normal operation there are additional wet cooling towers which are used to increase the heat rejection capacity. The IPE assumes that the wet cooling towers are needed in case of a LOCA, when several types of safety equipment may be operating simultaneously. The system which cools the CCW system and rejects the heat to the wet cooling towers is known as the auxiliary component cooling system (ACCW), and is only needed in case of LOCAs, as far as the IPE is concerned. This system has two pump trains and two wet cooling towers.

- The CCW is needed to cool the HPSI pumps, the LPSI pumps, containment spray pumps, shutdown heat exchangers (also used for containment spray recirculation cooling), containment fans, the emergency diesel generators and the central chillers used to provide HVAC cooling for several plant areas.

- The instrument air system is necessary for operation of the MFW system and the normal pressurizer spray (but not the auxiliary spray, supplied by the charging pump). All the other important systems (EFW, CCW, ACCW, containment sump recirculation valves) are provided with a backup air or nitrogen accumulator system. There are two instrument air compressors, of which one is sufficient to supply the requisite loads in an intermittent type of operation. In case of failure of both compressors, a cross tie to the station air system automatically opens; the station air has three compressors. Therefore the compressed air system seems to be relatively reliable and the systems affected are relatively few.

- Room cooling or ventilation is needed for several important systems: HPSI (not needed during the RWSP injection phase due to the low temperature of the water pumped), LPSI (not needed during the injection phase), containment sprays (not needed in the injection phase), MDEFW pumps, normal pressurizer sprays, emergency diesel generators and the CCW pumps.

- The switchover to recirculation is automatic. However, the operator must manually close the RWSP (refueling water storage pool) suction valves at that time.

- The recirculation spray (using the CSS pumps aligned to the containment sump and the shutdown heat exchangers) is necessary to provide cooling of the containment sump water.

- LPSI is automatically stopped on switchover to recirculation and HPSI is automatically aligned to the sump (along with the CSS) even if a LPSI operated in the injection mode, and even though LPSI path for recirculation (through the shutdown heat exchangers) exists. The reason is that the LPSI pumps may cavitate when simultaneously taking suction from the containment sump with the containment spray pumps. Since the IPE, a hardware modification has been implemented such that the LPSI pumps can be used to provide the recirculation spray in case of failure of the spray pumps.

2.5.2 Other GSIs/USIs Addressed in the Submittal

No other USIs and GIs are addressed in the submittal.

2.5.3 Response to CPI Program Recommendations

The CPI recommendation for PWRs with a dry containment is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements. Although the effects of hydrogen combustion on containment integrity and equipment are discussed in the submittal, the CPI issue is not specifically addressed in the submittal. More detailed information on this issue is provided in the licensee's response to the RAI (Level 2 Question 13). According to the response, although no containment walkdowns were conducted specifically for Level 2, the Waterford 3 PSA staff has made many trips into the containment and has a good understanding of the geometry of the containment.

According to the response, the Waterford 3 containment is a very open design that is not compartmentalized, and with the possible exception of the reactor cavity, all parts of the containment atmosphere are expected to be well mixed during an accident scenario. The reactor cavity is the only relatively enclosed volume in the containment. Since the reactor cavity volume is surrounded by thick reinforced concrete walls sized to withstand a large break LOCA blowdown and since no equipment is located in this area, hydrogen combustion in the cavity is not expected to affect any safety significant equipment. Additionally, according to the response, hydrogen detonation is not believed to be likely in the Waterford 3 containment. As can be seen in the above description, the discussions provided by the licensee on this issue is qualitative in nature, no quantitative information is provided in the discussion.

2.6 Vulnerabilities and Plant Improvements

The vulnerability criteria used for the IPE by the licensee are as follows:

1) A mean core damage frequency of 1.E-4/yr or greater for any sequence.

2) A sequence that contributes more than 50% to the total CDF.

3) A single failure or a common cause failure or an operator failure which has an unusual or significant effect on the CDF.

4) A support system failure which causes multiple frontline system failures and thereby has an unusual or significant effect on the CDF.

Based on these criteria no vulnerabilities were found.

The IPE did not take credit for any potential improvements. The potential improvements shown below (except for LPSI employment for recirculation spray which has been implemented) have not been evaluated yet, but are scheduled for disposition within the framework of the severe accident management guideline preparation effort, scheduled for completion by summer of 1997.

No impact on the CDF of any improvements has been evaluated.

The following are the improvements considered as a result of the IPE:

Hardware:

1) Install a portable generator to charge the AB battery. This will reduce SBO contribution from depletion of this battery which is used to control the TDEFW pump.

2) Provide feedwater from the fire protection system to the steam generator. The fire protection system has its own diesel driven pumps. During SBO or total loss of feedwater, this system could be used provided the SG were depressurized to below 200 psia, the shutoff head of these pumps.

Operating procedures:

1) Provide additional chiller/HVAC failure guidance. Room cooling is important as a contributor to the CDF and because it cools HPSI and EFW (MD) pumps. The failures are typically slow acting so the operators have time to respond. Therefore additional guidance may insure a timely response.

2) Cross-tie of AC power trains. Proceduralize the cross-tie between the A and B trains (hardware already exists). Drills have demonstrated the pertinence of this type of recovery. A procedure will make it easier to accomplish it in a shorter time.

3) Enhance refill of the CSP. CSP drawdown is an important contributor. Emphasizing the need to monitor level and makeup from the wet cooling tower basins or the CST will help prevent this from being a contibutor.

4) Add guidance for aligning LPSI pump for containment spray. Containment cooling is needed in the recirculation phase to insure NPSH of recirculation pumps. Hardware connections exist for LPSI to take over the recirc spray function in case of CSS pump failure, however, currently, LPSI pumps are disabled from recirculation. This is because they would cavitate if operated together with the CSS pumps to take suction from the containment sump. Since in this case CSS pumps are not available, LPSI pumps can take over to provide CHR. This procedure guidance has already been implemented.

The following changes were made in response to the SBO rule:

1) Stripping of DC loads was added to the procedure for SBO coping. This stripping allows the AB battery to last 4 hours. This was credited, but not analyzed in the IPE (i.e., no HEPs assigned);

2) All three safety batteries were replaced with batteries of increased capacity to allow for aging. This was not credited;

3) A new non-safety battery was installed in the turbine bldg to remove non-safety loads from the AB battery. This change greatly increases the duration of the AB battery in SBO conditions. This was not credited (came after IPE submittal);

4) Thermometers were installed in certain plant areas to confirm the initial temperature assumed in SBO calculations. This change does not affect the IPE;

5) A diesel power air compressor was installed which allows recharging of the EDG air starting system. Starting air can be supplied to restart the EDG if previous starting attempts have exhausted the compressed air supply from the starting system.

No CDF impact of these changes is evaluated.

The following additional back end potential plant improvements are discussed in the IPE submittal:

1. Enhance communication between sump and cavity -- A hardware change (e.g., removal the door in the cavity cooling duct work) may be performed to increase the flow of the water in the containment sump to the reactor cavity.

2. Provide water from the fire protection system to the containment sump -- This can Provide water to the reactor cavity to prevent vessel breach by allowing ex-vessel cooling.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

Based on the Level 1 review of the Waterford 3 IPE the licensee appears to have analyzed the design and operations of Waterford 3 to discover instances of particular vulnerability to core damage. It also appears that the licensee has: developed an overall appreciation of severe accident behavior; gained an understanding of the most likely severe accidents at Waterford 3; and implemented changes to the plant to help prevent and mitigate severe accidents. It is not clear that quantitative understanding was gained by the licensee due to a number of data problems (see below).

Strengths of the Level 1 IPE are as follows: Thorough analysis of initiating events and their impact, descriptions of the plant responses, modeling of accident scenarios, generally reasonable failure data and common cause factors employed and usage of plant specific data where possible to support the quantification of initiating events, diesel generator failures and component maintenance unavailabilities. The flooding analysis seems to have been reasonable and thorough. The effort seems to have been evenly distributed across the various areas of the analysis. The documentation was usually good, and reasonable effort was made to provide RAI responses. Some pessimistic assumptions were employed to offset some of the optimistic aspects of the analysis.

The weaknesses were in using seemingly low values for some important data: LOOP and small LOCA initiating event frequencies, power recovery curve, somewhat low CCF for MDEFW pumps and omission of some CCFs. The TDEFW run failure number is low compared to the NUREG/CR-4550 recommended value. Shedding of DC loads was not modeled. There is uneven modeling of common cause failures and some common cause failures are omitted from the analysis: It is not clear if CCF of all three HPSI pumps or all three CCW pumps was considered. HVAC modeling of the shutdown heat exchanger room is not clear. These comments may have a moderate to large (in case of power recovery factors) impact on the results. However, they may be somewhat offset by some pessimistic assumptions: EDG run failure occurs at the beginning of the SBO, no credit for TDEFW operation with water at inlet, large maintenance unavailaiblity of the dry cooling tower, and no credit for recent battery upgrades such that low shedding may not be required.

The IPE determined that failures in the AC power, EFW, ACCW, HPSI, CCW and HVAC dominate the risk profile. Loss of offsite power and small LOCA account for about 80% of the total CDF. SBO accounts for about 38% of the CDF. The CDF is dominated by 5 accident sequences (not accounting the ISLOCA which contributes about 3%).

The HRA review of the Waterford 3 IPE submittal and a review of the licensees responses to HRA related questions asked in the NRC RAI, revealed several weaknesses in the HRA as documented. In general, a viable approach (the Dougherty and Fragola method) was used in performing the HRA, but several weaknesses in how the analysis was conducted (or at least in the licensees documentation of the conduct of the analysis) were identified. While the weaknesses are not severe enough to conclude that the licensees submittal failed to meet the intent of Generic Letter 88-20 in regards to the HRA, they do suggest the licensee may not have learned as much about the role of humans during accidents as would have been possible. Important elements pertinent to this determination include the following:

1) The submittal indicates that utility personnel were significantly involved in the HRA. Regarding the IPE HRA representing the as-built, as-operated plant, the submittal states that "the HRA task

served as an integral advisor to other project tasks to assure that relevant human interactions were identified and properly incorporated into the logic models." The HRA task was involved during initial sequence and modeling efforts and "during this period had the opportunity to review plant and system design information and become familiar with the control room and related operating procedures." While simulator exercises were not conducted, the statements discussed above suggest that the HRA analyst was significantly involved throughout the modeling effort. Thus, it appears that steps were taken to assure that the HRA represented the as-built, as-operated plant. However, documentation of HRA related walkdowns and observations of simulator exercises would have strengthened the notion that a viable process was used.

2) The submittal indicated that the analysis of pre-initiator actions included both miscalibrations and restoration faults. An acceptable, but potentially optimistic analysis was conducted Events found to be potentially risk significant were analyzed in detail using an "SAIC" method that is "a variant on THERP and is similar to the ASEP HRA procedure.

3) The major limitation of the post-initiator analysis concerns the extent to which plant-specific factors were considered. While the model itself provides reasonable mechanisms for addressing relevant plant - specific factors, on the basis of examples provided, it would appear that many of the parameters were left at their default values and that potential PSFs were not carefully considered. The resulting analysis therefore appears to be "generic" rather than plant-specific and may or may not adequately represent the plant.

4) Consideration of dependencies between separate tasks was essentially treated by assuming that they are independent. The licensee argues that "between separate tasks independence is provided because many of the tasks are performed by different people, and there is separation in time or "cognitive space", i.e., cues are independent enough to force subsequent diagnosis." The licensee further states "that context effects were handled by lumping the different sequences into one event." "This is done by using a sum average time for the available time parameter for events that are sequence dependent." These statements apparently reflect a "bounding" approach that could lead to pessimistic or optimistic HEPs, depending on the circumstances.

5) A list of important human actions based on their contribution to core damage frequency was provided in the submittal.

6) The HRA portion of the flooding analysis appeared reasonable and thorough..

The IPE uses small containment event trees (CETs) for Level 2 analysis. The quantification of the CET in the Waterford 3 IPE is based on review of industry literature and plant-specific calculation using the MAAP code.

The interface between the Level 1 and Level 2 analyses is accomplished by the development of a set of 10 plant damage states. The Level 1 core damage sequences are grouped in the plant damage states based on RCS pressure, core melt timing, and the availability of containment mitigating systems. Separate CETs are used for bypass PDSs, SBO PDSs, and other PDSs. The definition of the PDSs for the Level 1 and Level 2 interface seems adequate. The CETs used in the IPE provide a reasonable coverage of the important back-end phenomena. The quantification of the CETs also seems adequate.

The important points of the technical evaluation of the Waterford 3 IPE back-end analysis are summarized below:

- The back-end portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20.

- The Waterford 3 Steam Electric Station IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.

- The high early containment failure probability obtained in the Waterford IPE submittal is partially due to the use of a conservative containment fragility curve. The conditional early failure probability is reduced from 26% to 4% when a revised fragility curve more consistent with those used in other IPEs is used.

- Despite the use of a 24 hour mission time late containment failure occurs if both containment spray and containment fan coolers fail. On the other hand, because of the use of a mission time, the probabilities of containment failure by noncondensable gases and basemat melt-through are assumed to be low even if the debris is not coolable.

- The IPE has identified a plant-specific reactor cavity configuration feature that may affect accident progression. Based on the IPE, it is recommended that the communication between sump and cavity be enhanced. This may be achieved by removing the door in the cavity cooling duct work to increase the flow of the water in the containment sump to the reactor cavity.

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