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TECHNICAL EVALUATION REPORT OF THE IPE SUBMITTAL AND RAI RESPONSES FOR THE PRAIRIE ISLAND UNITS 1 AND 2 NUCLEAR GENERATING PLANT

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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 Prairie Island Nuclear Generating Station. The primary intent 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 (RAI Responses) provided by the licensee, the Northern States Power Company, in the response to a request for additional information (RAI) by the NRC.

E.1 Plant Characterization

The Prairie Island Nuclear Generating Plant consists of two 560 MWe two-loop Westinghouse pressurized water reactor units, Unit 1 and Unit 2. The reactor coolant system (RCS) of each unit includes the pressure vessel, two vertical steam generators, two reactor coolant pumps, an electrically heated pressurizer and interconnected piping. The RCS is housed inside a large dry containment. Reactors with similar characteristics are: Ginna, Kewaunee and Point Beach 1 and 2.

The plant is located near Red Wing, Minnesota. It is operated by the Northern States Power Co. (NSP). Full commercial operation began on December 16, 1973 for Unit 1 and December 21, 1974 for Unit 2.

A number of design features at Prairie Island 1 and 2 impacts the core damage frequency (CDF). The submittal highlights these features, but does not estimate their effects on the CDF individually.

The following features tend to decrease the CDF:

- The emergency AC power configuration includes four diesel generators of diverse design and support system requirements. In the event of an SBO condition, each diesel generator has the capability to supply the power requirements for the hot shutdown loads for its associated unit, as well as one train of essential loads of the blacked out unit through the use of manual bus tie breakers interconnecting the 4160 buses between units. The crosstie of a safeguards 4160 bus from one unit to the diesel generator for the other unit can be performed from the control room, within ten minutes of the onset of SBO.

- The cooling water system is diverse. It consists of five pumps of which two are horizontal motor driven, and three are vertical pumps with one motor driven and two diesel driven. The single vertical motor driven pump is also backed by safeguard diesel generators (before the SBO rule the pump was powered by a non safeguards power supply) while the two diesel driven pumps do not rely on AC power for operation, i.e., there are three cooling water pumps available following a LOOP. The cooling water system consists of a ring header that can be divided into two separate headers on receipt of an SI signal. Each header supplies half of both trains of safeguards equipment.

- The secondary cooling function involving the main feedwater and auxiliary feedwater systems is very reliable. Main feedwater is lost on a reactor trip but is easily recoverable from the control room. The feedpumps are motor driven, i.e., independent of main steam availability. Each unit has a turbine and a motor driven auxiliary feedwater pump. The motor driven AFW pumps can

be crosstied between both units. Large condensate storage tanks (CST) are capable of providing water for decay heat removal without the need for makeup.

- The equipment located in the Auxiliary Building does not require room cooling for extended periods of operation. Analysis has been performed which demonstrates that the CS, CC, RHR and SI pumps do not require ventilation for sustained periods of pump operation.

- The RWST is large and can provide many hours of makeup to the reactor for small break LOCA and SGTR.

The following features tend to increase the CDF:

- Both main feedwater and feed and bleed cooling are dependent on common support systems; instrument air, cooling water and DC power.

- Cooling water, instrument air and control room chilled water systems are shared by both units. These are systems that are required to be fully operational when either unit is at power. Maintenance on these systems is normally performed while both units are at power. Maintenance activity may influence negatively the availability of these systems.

- The auxiliary feedwater pumps for both units are all located in the same room such that a pipe rupture in the loop A or B cooling water line can result in the failure of all auxiliary feedwater for both units.

- The instrument air compressors are also located in the same room as the auxiliary feedwater pumps such that all the compressors could fail because of a flood, causing loss of instrument air for both units. Loss of instrument air results in closure of the main feedwater regulating and bypass valves which together with auxiliary feedwater failure, results in the loss of secondary cooling. Feed and bleed then fails because the pressurizer PORVs require instrument air to operate.

- The emergency batteries have only two hours of capacity.

- Loss of instrument air will cause the control room chiller outlet cooling water valves to close resulting in loss of chilled water and loss of room cooling to the unit 1 480 V safeguards bus rooms. Without operator intervention, the rooms can heatup and fail the 4160/480 V transformers resulting in a loss of all unit 1 480 V safeguards equipment. This would cause loss of all charging pumps resulting in a loss of all RCP seal cooling causing an RCP seal LOCA in which the SI system would not be available for mitigation.

- Both RCP seal cooling and RCS short term inventory control are dependent on cooling water. Cooling water provides the ultimate heat sink for the component cooling water system which provides cooling to the RCP thermal barrier and the SI pump lube oil coolers. Cooling water also supplies a heat sink for the control room chillers which provide room cooling for the Unit 1 safeguards 480 V bus rooms. On loss of cooling water, room cooling is lost to the Unit 1 safeguards 480 V bus room. Without operator intervention the room can heatup and fail the

4160/480 V transformers resulting in, as previously mentioned, a loss of all Unit 1 480 V safeguards equipment.

According to the licensee's response to the RAI, since the IPE was submitted room ventilation problems involving the control room chilled water system have changed. Each of the 480 V buses was split into two and now each of the buses carries half of the original loads. In addition, only one of these buses is now located in a room. Furthermore, new thermal analysis showed that without ventilation and assuming no operator action, these room temperatures do not reach debilitating levels within 12 hours. With operator action (to open doors at 12 hours) room temperatures decrease to a steady-state temperature well below operability limits.

- The instrument air system has a high failure probability as the system success criterion is such that if two out of three compressors fail, instrument air is considered failed. A single compressor cannot maintain adequate header pressure for both units.

- Given a medium LOCA, switchover to high head recirculation can not be performed from the control room as the RHR to SI crossover motor valves have their breakers locked in the open position. The switchover to high head recirculation must be accomplished within a small time window during which both the SI, CS and/or the RHR pumps are injecting from the RWST. If the operator fails to stop any of the pumps before the RWST level decreases below approximately 5%, all pumps will be damaged as they do not have suction trips.

For additional unique features related to the CDF see Section 1.2 of this report.

The Prairie Island Nuclear Generating Station utilizes a large dry containment with a freestanding steel shell construction. The steel reactor containment vessel is enclosed by a 2-1/2 foot thick concrete shield building. The reactor coolant system is a Westinghouse two-loop design. Both the power level and the containment free volume of Prairie Island are significantly less than those of Zion Surry. However, the containment volume to thermal power ratio for Prairie Island is comparable to, and between that of Zion and Surry. The containment design pressure of Prairie Island is between that of Zion and Surry, and the median containment failure pressure obtained in the Prairie Island IPE is slightly higher than that obtained in NUREG-1150 for Zion and Surry.

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 upper compartment to the annular containment floors. Flooding of the cavity is accomplished through an access hatch (located on the instrument tunnel) which is left open (slightly ajar) during normal operation and allows water flow from the containment floor down into the cavity for vessel head flooding. This provides external cooling to the core inside the reactor vessel. Since the flow velocity going into the instrument tunnel may be sufficient to pull the doors closed, a recommendation of securing the hatches open by installing a solid bar or other device, instead of a chain, is discussed in the IPE submittal.

- A steel shell containment that is vulnerable to direct attack by dispersed core debris. The access hatches to the instrument tunnel are in an open area on the basement level of the containment, and for both of the Prairie Island units one of the two hatches faces toward the steel containment, about 30 ft away, with a largely unobstructed path in between. Although a scoping study

performed in the IPE shows that the temperature generated by the debris adhering to the steel wall is insufficient to melt the steel and breach the containment, details of the scoping study are not provided in the submittal and the potential effect of corium attack on reducing containment pressure capability is not discussed.

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

- Two separate systems for containment atmosphere cooling and pressure suppression, the Fan Coil Units and the Containment Spray system. According to the IPE, the low conditional probability of containment failure for core damage sequences that do not bypass containment is due in part to the availability of these two completely redundant, diverse means of providing containment heat removal and pressure suppression.

- An Emergency Procedure (EP) that requires the restart of the reactor coolant pumps (RCPs). According to the IPE, the Emergency Procedure "Response to Inadequate Core Cooling" creates the possibility of inducing a steam generator tube rupture (SGTR) during an event in which degraded core cooling conditions already exist. A recommendation of revising the EP is discussed in the IPE submittal.

E.2 Licensee's IPE Process

The licensee initiated work on a probabilistic risk assessment (PRA) for Prairie Island Nuclear Generating Station in response to Generic Letter 88-20. The freeze date for the analysis is 1990.

The front-end portion of the submitted IPE is a Level 1 PRA. The specific technique used to develop this PRA was the small event tree/large fault tree approach, and it is clearly presented in the submittal.

Internal initiating event and internal flooding were considered. Event trees were developed for all classes of initiating events. The details of the technique applied is extensively described in the submittal: Support systems were modeled in the fault trees of the functional top events and accident sequences were solved by fault tree linking. For deterministic best estimate analysis of reactor and containment response during severe accident sequence condition the MAAP 3.0B Revision 19 code was used as principal calculational tool. Mission time for logic model quantification were generally on the order of 24 hours. The consequences of system and equipment failure that might occur during this period were examined well beyond this mission time. Containment response and source term analyses were carried out at least 48 hours to establish important trends in plant response, timing and magnitude of potential releases that might occur beyond 48 hours were established based on these trends where necessary.

Sensitivity studies were conducted on initiating event frequencies, operator actions, common cause, test and maintenance and for certain system components. The evaluations were performed to determine the global effect of the parameters of interest. Failure rates were increased/reduced by a factor of 5, where a higher level of uncertainty and variability exists such as human reliability, common cause and test and maintenance unavailability. Failure rates for system components were increased/decreased by a factor of 2 since actual plant data were used and there was less uncertainty associated with these parameters. The licensee did not evaluate explicitly the changes in CDF due to credited SBO modifications.

In addition to the sensitivity studies, importance analyses were performed on the same quantities. The importance analyses were based on use of the Fussel-Vesely and Birnbaum algorithms.

Sensitivity analyses were conducted also to estimate the reduction in CDF due to certain plant improvements (AFW room fire door closed and procedural changes to allow station air to be cross -tied with instrument air).

The submittal states that the Northern States Power (NSP) PRA staff was involved in all aspects of the PRA. Three of the team members were located at the general office and two were located on site to "make it easier to interface with plant staff and to conduct walkdowns to ensure the PRA represents the as built plant." In addition, procedure reviews, discussions with operations and training staff, and walkdowns of some operator actions helped assure that the IPE HRA represented the as-built, as-operated plant. Contractors associated with the Individual Plant Evaluation Partnership (IPEP) supported the PRA. Four levels of review were conducted and they included both internal and external reviews. However, a specific review of the HRA was not discussed. 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. Important human actions were identified and several recommendations related to improving procedures and operator training were discussed in the submittal.

The back-end containment analysis was performed by Northern States Power Company (NSP) with the help of TENERA, Westinghouse Electric Corporation, and Fauske & Associates Inc.. In addition to the above companies, Gabor, Kenton & Associates (GKA) also provided consulting services to NSP. According to the IPE submittal, the NSP PRA staff was involved with all aspects of the IPE. Reviews performed in the Prairie Island IPE include an independent in-house review done by NSP personnel other than those on the NSP PRA staff. The review process used in the Prairie Island IPE as described in the submittal seems to satisfy the intent of Generic Letter 88-20.

E.3 IPE Analysis

E.3.1 Front-End Analysis

The IPE initiating event analysis used the work performed previously in the IDCOR IPEM, but significantly enlarged its scope and employed updated data. The total number of initiating events is 23. They are grouped into seven categories (the internal flooding events and anticipated transients are classified together and the different ATWS events are collapsed into one category), since it was found that the minimal number of significantly different ways in which the plant responds to challenges is seven.

Transient occurrence data (from the period 1/1/80-12/31/90) were used to derive the plant-specific initiator frequencies. Generic initiating event frequencies were obtained from published sources. The SGTR initiator frequency was obtained by Bayesian update. Plant-specific system fault tree models were used to estimate the special transient initiating event frequencies (loss of DC Train A and B, loss of Instrument Air, etc.) and Interfacing systems LOCAs.

The IPE developed eight event trees to model the plant responses to internal initiating events: large, medium and small LOCA event trees, SGTR event tree, main steam line/feedwater line break event tree, transient event tree, loss of offsite power (including station blackout) event tree and ATWS event tree.

No event trees were developed for interfacing systems LOCAs but each ISLOCA source was separately analyzed.

No separate event trees were developed for flooding scenarios, the transient event tree was used with additional flood-caused failures flagged in the appropriate fault trees.

A total of 21 systems/functions are described in Section 3.2.1 of the Submittal. Included are descriptions of the following systems: reactor protection, AMSAC, CVCS, safety injection, RHR, main feedwater and condensate, auxiliary feedwater, pressurizer PORV, containment fan coil units, containment spray system, cooling water, component cooling water, instrument air, onsite AC power, DC power, 120 V instrument AC power, SI signal, safeguards chilled water, SG PORV, MSIV, room cooling.

Each system description includes a discussion of the system design and operation, dependencies, success criteria and role in safety function(s).

Importance analyses (by using Fussel-Vesely and Birnbaum importance indicators) were performed for the initiating events, the major operator actions and all the systems considered in the plant risk model. The importance analysis also included the corrective maintenance contributions as well as the preventive maintenances and test contributions to the overall CDF.

Sensitivity analyses were conducted on initiating event frequencies, operator action failure rates, common cause, test and maintenance and for certain system components (e.g., EDG failure rates).

Common cause failures of circuit breakers, switchgear and relays were not explicitly modeled. According to the RAI response they were considered implicitly. For instance, common cause failures of loads supplied through the breakers, such as pumps, valves and other components that can be attributable to common cause mechanisms, were modeled.

The Prairie Island IPE team systematically examined the plant model for redundant components to address potential common-cause failures (within individual systems and across both units). The component groups for which common cause events were defined are given below:

1. Diesel generators (failure to start and run),
2. Pumps (failure to start and run),
3. MOVs and AOVs (failure to open and close),
4. PORVs (failure to open or reclose on demand),
5. Check Valves (failure to open on demand; failure to reclose),
6. Batteries (failure to operate on demand),
7. Instrumentation and Control components (failure to send signal or actuate equipment),
8. Air compressors (failure to start and run),
9. Cooling fans (failure to start and run),
10. Chillers (failure to start and run).

The common cause probability model used was the Multiple Greek Letter (MGL) method.

The methodology of the IPE to perform the flooding analysis consisted of three major steps:

1) Identification of potential floods and areas affected (flood zones),

2) Identification and initial screening of flooding scenarios, and
3) Quantification of important flooding scenarios.

In addition, extensive plant walkdowns supported the development of the flooding scenarios.

After a thorough screening process, six flood zones were retained for more detailed analysis.

To quantify the flood initiating event frequencies the EPRI document TR-102266 was used and several assumptions regarding systems and equipment that may be disabled as a result of the flood or as a consequence of operator actions have been made.

The total contribution of internal flooding to the point estimate CDF was estimated to be 1.04E-05/yr, which is about 21% of the total CDF (from internal events and internal flooding). This is dominated by a single flood scenario, that accounts for almost all of the CDF due to flooding. The flood is in zone TB1 which is the Auxiliary Feedwater Pump/Instrument Air Compressor Room. This room has the main cooling water supply headers to the Auxiliary Building running through the overhead.

The IPE point estimate for the core damage frequency from internal events and internal flooding is 5.0E-05/yr for Unit 1 and 5.1E-05/yr for Unit 2. Accident initiators and their percent contribution to the CDF for both units, Unit 1 and Unit 2 are listed in Table E-1. As can be seen, the core damage contributions by initiating event of Unit 2 closely resemble to those of Unit 1. This is because there are only few and minor asymmetries in the designs and corresponding risk models of the units and the Unit 2 Level 1 results were obtained essentially from requantifying the Unit 1 model and replacing appropriate Unit 1 component failures by their counterpart for Unit 2.

Table E-2 shows the CDF results in term of grouped accident types and their percent contribution. The largest contribution is due to the transient group (including the LOOP events). LOCA and internal flood events are the next largest contributors and they contribute almost equally. The SGTR's contribution is quite significant.

The CDF results were obtained in the form of functional sequences.. Therefore the submittal used those screening criteria for reporting, which were required for such sequences in Generic letter 88-20. In Table 3.4-1 of the submittal all the accident classes (including those that meet the reporting criteria and those that are beyond them) constituting the total CDF are presented (altogether 15 accident classes). The table contains also the description of the dominant sequence from each accident class (except accident class "TEH", from which two leading sequences were considered). These dominant sequences cover only 36.4% of total CDF. The sequences include: five LOCAs, two SGTRs, six transients, two floodings and one interfacing LOCA.

The importance analyses performed on the initiating events shows that the most important initiators at Prairie Island are: the LOOP(SBO) events, the Floods (T1FLD, SH1FLD), the LOCAs (MLOCA, SLOCA, LLOCA) and the LOCL (loss of cooling/service water) event.

The importance analysis on the systems provided the ranking of the systems contributing the most to the total CDF as follows: 1) the AFW, 2) AC power, 3) the room-cooling, and 4) the cooling water systems. First is the AFW, because in many accident sequences due to a variety of initiators (e.g.; LOOP, T1FLD, Loss of IA, Loss of DC) it is the only remaining means of secondary cooling. AC power is in second place; it causes heavy reliance on the TDAFW pump during an SBO when the MFW

and MDAFW pumps are lost. Room cooling is important, because when Loss of IA or Loss of CL occur, the chilled water, that supplies the room cooling of the Unit 1 480 V safeguards bus rooms, fails. (This condition is significantly ameliorated, as discussed earlier in the present TER based on the RAI response.)

Table E-1 Core Damage Frequency by Initiating Event

Initiating Event	CDF from Initiating Event (per reactor year) Unit 1	% of Total CDF from Initiating Event Unit 1	CDF from Initiating Event (per reactor year) Unit 2	% of Total CDF from Initiating Event Unit 2
I-TR1	6.4E-07	1.3	6.6E-07	1.3
I-TR2	2.9E-08	0.06	3.1E-08	0.06
I-TR3	1.2E-06	2.4	1.2E-06	2.4
I-TR4	5.2E-07	1.0	5.5E-07	1.1
I-LOCC	5.5E-07	1.1	5.5E-07	1.1
I-LOCL	6.4E-07	1.3	6.4E-07	1.3
I-LOCA	2.2E-06	4.4	2.2E-06	4.3
I-LOCB	4.6E-07	0.9	4.8E-07	0.9
I-INSTAIR	3.2E-06	6.3	3.2E-06	6.2
I-LOOP	1.1E-05	21.2	1.1E-05	22.4
I-MSLB	*	*	*	*
I-MFLB	*	*	*	*
I-SLOCA	4.1E-06	8.2	4.2E-06	8.2
I-MLOCA	4.6E-06	9.3	4.6E-06	9.1
I-LLOCA	3.7E-06	7.5	3.8E-06	7.3
I-SGTk	6.6E-06	13.2	6.6E-06	13.0
I-T1FLD	1E-05	21	1.04E-05	20.4
I-T13FLD	*	*	*	*
I-AB7FLD	8.5E-10	2E-03	1.5E-09	2.9E-03
I-AB8FLD	*	*	*	*
I-SH1FLD	4.1E-07	0.8	4.1E-07	0.8
I-SH2FLD	4.3E-10	9E-04	5.6E-10	1.1E-03
V	2.3E-07	0.5	2.27E-07	0.5

Table E-2 Accident Types and Their Contribution to the CDF

Initiating Event Group	Contribution to CDF (/r. yr)	%
LOCAs	1.22E-05	24.3
Steam Generator Tube Rupture	6.6E-06	13.2
Interfacing System LOCA	2.27E-07	0.5
Transients (w/o LOOP)	9.44E-06	18.8
Anticipated Trans. w/o Scram	3.2E-07	0.6
Internal Flooding	1.04E-05	20.7
LOOP (with SBO)	1.1E-05 (3.1E-06)	21.9 (6.2)
TOTAL CDF	5.02E-05	100.0

E.3.2 Human Reliability Analysis

The HRA process for the Prairie Island 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 considered both miscalibrations and restoration faults, but only restoration faults were explicitly modeled. The potential for miscalibration of a group of sensors or instruments was assumed to be included in the common cause failure modeling of the instruments themselves. With the exception of events excluded on the basis of a qualitative screening applied during the pre-initiator human action selection process, all pre-initiator restoration errors were quantified. One of two HEPs (0.003 or 0.01) was assigned to each event depending on whether or not a post-maintenance verification of component status was required by procedure. Plant-specific component unavailabilities were then calculated using the HEPs in conjunction with factors such as maintenance frequency and duration, test frequency and interval, refueling outage frequency, and time from completion of the corrective maintenance to the retest of the component.

Post-initiator human actions modeled included both response-type and recovery-type actions. The submittal indicated that screening values were assigned "by following a flow chart and answering a series of questions." Neither the questions nor the flow chart was provided, but "plant-specific estimates of the time available to initiate and perform the action" and "PSFs associated with degree of difficulty and stress" were considered. The screening HEPs were based on the Handbook (NUREG/CR-1278), Wash-1400 (NUREG-75/014), and "the data sources used in the IDCOR BWR IPE methodology." Post-initiator response type actions found to be important received detailed HEP development. An action was important if it contributed significantly to baseline core probability or "if a change in the failure rate could cause significant increase in overall core damage probability." Most of the actions receiving detailed analysis were quantified with the ASEP method (NUREG/CR-4772). The response to the NRC RAI states that a more refined estimate for the five most important human actions was obtained using the Handbook (NUREG/CR-1278). The licensee's response to the RAI also indicated that most recovery probabilities were derived from NSAC-161, "Faulted Systems Recovery Experience" and that recovery

actions were only added to cutsets when it was apparent that an operator would have sufficient time to perform the additional action. Local recovery of valves which failed to open or close was credited only when there was control room indication of valve position and the valve was easily accessible. A non-recovery probability of 0.025 was assigned to these events. Recovery of equipment such as pumps had the same criteria, but the non-recovery probability was "approximately 0.5." Plant-specific performance shaping factors and dependencies (such as those among multiple actions in a sequence) were considered for both response and recovery actions. Human errors were identified as important contributors in accident sequences leading to core damage and several recommendations to improve procedures and operator training were provided.

E.3.3 Back-End Analysis

The Approach Used for Back-End Analysis

The methodology employed in the Prairie Island IPE for the back-end evaluation is clearly described in the submittal. Five containment event trees (CETs) are developed for the 14 accident classes (ACs) obtained in the IPE. Similar to plant damage states (PDSs) in other IPEs, accident classes are used in the Prairie Island IPE to group the core damage sequences obtained from the Level 1 analysis. However, unlike the PDSs used in the other IPEs, the availability of containment systems are not explicitly included in the definition of the ACs. In the Prairie Island IPE, containment system fault trees (for containment spray injection, containment spray recirculation, and containment fan coil units) were quantified as frontline systems along with the Level 1 frontline and support system fault trees using linked fault tree models. The containment systems fault tree cutsets were input to the CET branches as necessary to support CET quantification.

The CETs used in the Prairie Island IPE provide a structure for the evaluation of most of the containment failure modes discussed in NUREG-1335. The containment failure modes that are assumed negligible and thus not included in CET quantification include those from containment shell melt-through, vessel thrust force (the rocket mode failure) and penetration failure due to degradation of sealing materials under harsh environmental condition. The containment failure modes that are considered as unlikely but are assigned small probability values include those from direct containment heating, in-vessel steam explosion, ex-vessel steam explosion, and hydrogen combustion. Containment isolation failure is also considered as unlikely but is evaluated in CET quantification using the data obtained in containment isolation analysis.

The quantification of the CET in the Prairie Island IPE is based on plant-specific phenomenological evaluations, which include modeling and bounding calculations (based upon experimental data), consideration of phenomenological uncertainties, and MAAP calculations.

The result of the CET analysis are grouped to 17 CET end states. Release fractions for these CET end states are determined by the analyses of representative sequences using MAAP computer codes.

For the Prairie Island Nuclear Generating Station IPE, the definition of the interface between Level 1 and Level 2 analyses is reasonable. The CET is well structured and easy to understand. CET quantification and source term grouping and quantification also seem adequate. The IPE process is in general logical and consistent with GL 88-20.

Back-End Analysis Results

The conditional probabilities of the ACs obtained in the back-end analysis for the various accident initiators are: 23% for small LOCA (2 ACs), 22% for transient (2 ACs), 19% for internal flooding (2 ACs), 16% for medium or large LOCA (2 ACs), 13% for SGTR (2 ACs), 6% for SBO (1 AC), 0.6% for ATWS (2 ACs), and 0.4% for ISLOCA (1 AC). In terms of individual ACs, the leading AC is one with transient initiator with early core melt and the RCS at high pressure (20% of CDF). This is followed closely by ACs initiated by internal flooding (19%) and small LOCA (17%), both with early core melt and high RCS pressure, an AC with medium and large LOCAs with early core melt and low RCS pressure (15%), an AC initiated by SGTR (11%), and the AC for SBO accident sequences (6%).

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

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

Containment Failure Mode	Prairie Island Nuclear Generating Station IPE++	Surry NUREG-1150	Zion NUREG-1150
Early Failure	0.8	0.7	1.4
Late Failure	22.6	5.9	24.0
Bypass	44.7	12.2	0.7
Isolation Failure	0.02	*	**
Intact	31.8	81.2	73.0
CDF (1/ry)	4.9E-5	4.0E-5	3.4E-4

++ The data presented for Prairie Island are based on Table 4.7-1 of the IPE submittal. About 30% of bypass is due to induced SGTR (ISGTR). However, according to the licensee's responses to the RAI, ISGTR is effectively precluded by a procedure change recommended by the Westinghouse Owners Group and implemented at Prairie Island. The probability of bypass failure would decrease, and intact containment increase, by 30% if ISGTR is eliminated.

* Included in Early Failure, approximately 0.02%

** Included in Early Failure, approximately 0.5%

As shown in the above table, the conditional probability of containment bypass for Prairie Island is 44.7% of total CDF. Most of it is from induced steam generator tube rupture (30%). Excluding ISGTR, containment bypass is primarily from SGTR (3.2%) with small contribution from ISLOCA (0.5%).

The conditional probability of early containment failure for Prairie Island is about 0.8% of total CDF. It is about equally contributed by internal flooding (28% of early failure), small LOCA (26%), transient (20%) and medium/large LOCA (20%) accident classes. The contribution from SBO sequences is only about 6%. The smaller contribution from SBO sequences is partly due to the low CDF of SBO sequences (about 6% of total CDF). Early containment failure is primarily due to hydrogen burn with vessel breach

at high pressure (74% of early failure CDF). The remaining 26% of early failure is also due to hydrogen burn, but with no vessel failure, primarily from medium/large LOCA sequences.

The conditional probability of late containment failure for Prairie Island is 22.6% of total CDF. More than half of this probability is from small LOCA sequences (59%), with most of the remaining coming from transient sequences (38%). On a conditional basis, 58% of small LOCA sequences results in late containment failure and 39% of transient sequences results in late containment failure. The contribution from SBO sequences is relatively low in comparison with that from small LOCA or transient sequences (2% to total late failure CDF and 7% conditional probability). The low failure probability for SBO is probably due to ac recovery. The time allowed for ac recovery is long for late containment failure.

For Prairie Island, late containment failure is primarily due to overtemperature/overpressure failure caused by lack of cooling to the debris dispersed outside of the reactor cavity (78% of late failure CDF). Because a significant amount of core debris is dispersed to outside of the reactor cavity only in high pressure vessel failure cases, the conditional probability of late containment failure is low for accident classes that involve low pressure vessel failure or no vessel failure (e.g., medium/large LOCA and the late core melt accident classes). Besides the above late failure mechanism, another important late failure mechanism is that associated with the loss of decay heat removal (22% of late failure). The contribution from basemat melt-through is small because core debris is assumed to be coolable if RWST water is injected to the containment.

Source terms are provided in the IPE for 17 CET end states using MAAP code calculation results. Four of the 17 source terms are for containment bypass, three each for containment isolation failure, early containment failure and no containment failure, and four for late containment failure. Source term definitions are based on MAAP calculations for 11 selected sequences. Sequence selection is based on the consideration of the dominant sequence in each end state and other factors that influence the source term results. The sequence selection and the assignment of release fractions for source term determination seem adequate.

Two types of sensitivity studies are performed in the Prairie Island 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, performed in the IPE to establish the sensitivity of the Level 2 analysis to uncertainties in the physical modeling of containment response and the source term. The issues investigated in the sensitivity studies of the first type include ex-vessel cooling, RCS depressurization due to hot leg creep rupture, and coolability of the core debris in the reactor cavity and in the containment upper compartment (due to debris dispersal). The parameters investigated in the sensitivity studies of the second type include those recommended in the EPRI Guidance Document for performing sensitivity studies with MAAP 3.0B, the augmentations to these recommendations provided in the NRC sponsored MAAP 3.0B code evaluation, and specific areas deemed important for Prairie Island. The sensitivity studies provided in the Prairie Island 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 USI A-45 on decay heat removal (DHR). Several methods of DHR are discussed, including secondary cooldown and depressurization (using either AFW or main feedwater providing the

steam generator makeup), feed and bleed (i.e., utilizing the SI pumps and pressurizer PORV), safety injection and recirculation cooling (as provided by the SI and RHR systems), and shutdown cooling (by the RHR operation) . In addition containment cooling is mentioned. The CDF contributions from each of the individual DHR methods were estimated neither in the submittal nor in the RAI response.

The submittal provides a fairly detailed description of each of the diverse DHR capabilities at the Prairie Island plant. The description includes a reiteration of several specific features of the systems involved in DHR and major modeling assumptions, a discussion of the effects of initiating events on the systems' unavailabilities, and a presentation of the important hardware failures and operator errors contributing to these unavailabilities.

The NRC defines two requirements in NUREG-1289 that have to be met by any system which performs DHR. These are:

1. Maintain sufficient water inventory in the RCS to ensure adequate cooling of the fuel.
2. Provide the means for transferring heat from the RCS to an ultimate heat sink.

In the IPE there are no core damage sequence that do not involve loss of either one or both of the two requirements. This fact lead the licensee to consider the loss of DHR to be synonymous with core damage. Since the overall CDF was found to be acceptably low (5 E-05/yr) and it was shown that there are several redundant and diverse means for DHR, (i.e., several of the DHR systems and operator actions would have to fail in combination to have a serious negative impact on the DHR capability), the licensee considered that it has fulfilled the "Shutdown Decay Heat Removal Requirements" of "Unresolved Safety Issue A-45," (USI A-45).

No GSIs or USIs, other than USI A-45 are addressed 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 (Level 2 Question 10). Hydrogen sources, the condition for hydrogen detonation, and the load for hydrogen deflagration are discussed in the response. According to the response, hydrogen detonation is highly unlikely to occur with the Prairie Island containment geometry and hydrogen concentration. The loading condition generated by hydrogen deflagration has been pessimistically treated in the IPE and found not likely to cause containment failure.

E.5 Vulnerabilities and Plant Improvements

The criteria used in the Prairie Island IPE to determine whether any vulnerability existed at the plant were:

1. Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRAs?

2. Is there adequate assurance of no undue risk to public health and safety?

The licensee states that neither of the above criteria lead to the identification of potential vulnerabilities for the plant. The IPE process demonstrated that; the accident classes contributing to the CDF are comparable with those calculated in PRAs of similar nuclear plants (indeed the comparison made in Table 2.4-1 of the submittal with Surry, Kewaunee, Point Beach supports this viewpoint), and the overall CDF itself is at an acceptably low level of 5E-05/yr. Therefore the licensee believes that there is adequate assurance of no undue risk to public health and safety.

The licensee states that while no vulnerability exists, as a result of the IPE, recommendations have been generated for plant improvements. These recommendations are partly implemented partly or only under consideration but by no means represent any definitive "NRC commitments". The recommendations focus on plant improvements in three areas, such as: procedural/administrative, structural and training enhancements.

Depending on further evaluation of potential benefits and practicality, the licensee expects a significant decrease in the overall CDF (a decrease of 1E-05/yr or greater). In particular, the risk contribution from internal flooding is anticipated to be reduced from 1E-05/yr to approximately 1E-08/yr.

A summary of the Level 1 related recommendations is provided below:

Procedural/Administrative Enhancements

1. The plant already proceduralized the process establishing crosstie from station air to instrument air in C34 AOP1, Rev.0.

   a) If the crosstie could be established within one hour after a CL Loop A break, feed & bleed or main feedwater cooling could be restored and core melt could be prevented.(The station air compressors are cooled from Loop B cooling water and are not affected by a break in the other loop.)

   b) The new procedure prescribes also that station air crosstie should be used when an IA compressor is in maintenance.

Present (November 28, 1995) disposition of the recommendation: a) C34 AOP 1, Rev. 4 incorporates this action (Step 2.4.6); b) C34, Rev 12 incorporates this recommendation.

2. The procedure C35 AOP1, Rev.2, "Loss of CL Water Header A or B" should be revised such that the crosstie between CL Loop A and B could be used. Two valves, one manual and an AOV have to be opened during 20 minutes to supply cooling water to the MFW pumps' lube oil coolers. The MFW pumps can conservatively operate w/o cooling water for about 20 minutes before possible pump damage.

Present disposition of the recommendation: See the next disposition.

Structural Enhancements

1. Constrain the impact of AFW pump room flooding by some simple measures. Evaluations are underway to determine the best long term solution. In the interim the following measures are

suggested; modify the side doors to promote water flow out of the room, or close the fire door between the two halves of the room and render the door to be "water tight".

Present disposition of the recommendation: The CL header piping was completely replaced with a piping with a 33% thicker wall during the November, 1992 dual-unit outage. The internal surface of the new pipe is coated with an epoxy coating to inhibit microbiologically induced corrosion (MIC). Also piping failure would be noticed by any personnel who periodically walk through these rooms.

Enhancement of Training

1. Explain the importance of the Feed & Bleed process. Put an emphasis to the operator actions that are necessary for success. It is expected that the training will result in marked reduction of the contribution of accident class THE to the overall CDF.

2. Explain the importance of crosstie between the MDAFW pumps. Emphasize the operator actions that are required for success.

3. Explain the importance of switchover to high and low head recirculation. Emphasize the operator activities that are required to success. The training may reduce significantly the CDF contribution of the accident class SLL.

4. Explain the importance of RCS cooldown and depressurization to terminate SI before ruptured SG overfill. Emphasize the operator actions that are required for success. It is expected that the training results in reduction of the CDF contribution of accident class GLH.

The following two recommendations are related to the back-end:

1. Revise the EOP that requires the restart of the RCPs under ICC condition. It is recommended in the IPE submittal that the operator checks for adequate steam generator level before attempting to start the RCP. This recommendation is intended to reduce the probability of ISGTR.

2. Secure open the in-core instrument tube hatches for both units to allow water to flow into the reactor cavity to provide cooling to the lower vessel head (i.e., ex-vessel cooling) and improve debris coolability in the reactor cavity.

E.6 Observations

The licensee appears to have analyzed the design and operations of Prairie Island Nuclear Generating Station 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 Prairie Island Nuclear Generating Station; gained a quantitative understanding of the overall frequency of core damage; and implemented changes to the plant to help prevent and mitigate severe accidents.

Strengths of the Level 1 analysis of the IPE are as follows: Thorough analysis of initiating events and their impact for both units of the plant, descriptions of the plant responses, presentation of the results of supporting MAAP calculations (Section 7 of the submittal), reasonable failure data and common cause

factors, usage of plant specific data whenever possible to support the quantification of initiating events and system unavailabilities, and an importance analysis on major variables impacting core damage. The effort seems to have been evenly distributed across the various areas of the analysis.

The IPE Level 1 analysis has no fundamental weaknesses. Its "leading" minor weakness is the lack of detail in the individual accident sequence descriptions. The text of the submittal everywhere indicated that a detailed analysis was done but only highlights were reported.

The IPE determined that an internal flooding sequence is the primary contributor to core damage. Its initiator is a cooling water line break inside either the Unit 1 or Unit 2 side of the Turbine Building AFW/IA compressor room, which causes failure of these two systems for both units. There is a low potential for this pipe rupture (only a small section of piping is involved),however, this initiating event represents a potentially important location dependency for several systems at the plant, since the AFW pumps (secondary heat removal) and the IA compressors (in Feed & Bleed support for pressurizer PORV) are located in the same room.

As was noted previously, several recommendations for plant improvements have been made as a result of insights obtained from the IPE, particularly to reduce the CDF contribution of the above mentioned flood accident. The CDF impact of these improvements is expected to be a CDF decrease on the order of 1E-05/yr or greater.

The HRA review of the Prairie Island IPE submittal did not identify any significant problems or errors. A viable approach was used in performing the HRA and nothing in the licensees submittal indicated that it failed to meet the intent of Generic Letter 88-20 in regards to the HRA. Important elements pertinent to this determination include the following:

- The submittal indicated that utility personnel were involved in the HRA and the procedure reviews, discussions with operations and training staff, and walkdowns of operator actions represent a viable process for confirming that the HRA portions of the IPE represent the as-built-as operated plant.

- The analysis of pre-initiator human actions focused on restoration faults. The HEPs assigned to the modeled restoration faults and the approach for computing the component unavailabilities appeared reasonable. Dependencies between restoration errors were not addressed because it was argued that plant practices regarding maintenance of separate trains assured the independence of restoration faults. Miscalibration errors were "treated through the inclusion of common cause failure modeling for the sensors or instruments themselves." The licensee's treatment of miscalibration events may have precluded identification of important pre-initiator events and is therefore a weakness of the HRA.

- Post-initiator human actions modeled included both response-type and recovery-type actions. Although the documentation for the screening analysis performed on post-initiators was minimal, it appeared that the screening analysis was relatively more "plant-specific" and detailed than many of those performed for other IPEs. In addition, the licensee stated that if more than one operator action occurred in the same cutset, "either independence of the human actions was confirmed, or a change was made to correctly model dependence between human errors." The licensee's response to an additional RAI regarding treatment of dependencies confirmed that they were

appropriately addressed. Moreover, the detailed quantification of important post-initiator operator actions and the quantification of recovery actions appeared sound.

- Plant-specific performance shaping factors (PSFs) and event timing, were appropriately considered.

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

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 Prairie Island Nuclear Generating 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 in-core instrument tube hatches for both units be secured open to allow water to flow into the reactor cavity to provide cooling to the lower vessel head (i.e., ex-vessel cooling) and improve debris coolability in the reactor cavity.

- The steel shell containment of Prairie Island may be vulnerable to direct attack by dispersed core debris. The access hatches to the instrument tunnel are in an open area on the basement level of the containment, and for both of the Prairie Island units one of the two hatches faces toward the steel containment, about 30 ft away, with a largely unobstructed path in between. Although a scoping study performed in the IPE shows that the temperature generated by the debris adhering to the steel wall is insufficient to melt the steel and breach the containment, details of the scoping study are not provided in the submittal and the potential effect of corium attack on reducing containment pressure capability is not discussed.

- The IPE identified the potential of ISGTR due to the restart the RCPs upon an inadequate core cooling (ICC) condition. Based on the IPE results, it is recommended in the IPE submittal that the EOP that requires the restart of the RCPs under ICC condition be modified. It is recommended in the IPE submittal that the operator checks for adequate steam generator level before attempting to start the RCP. This recommendation is intended to reduce or eliminate the probability of ISGTR.

- The containment analyses indicate that there is a 68% conditional probability of containment failure if ISGTR due to restart of the RCP is included. The conditional probability of containment bypass is about 45% of which 30% is from ISGTR, the conditional probability of early containment failure is 0.8%, the conditional probability of isolation failure is about 0.02%, and the conditional probability of late containment failure is 23%. The conditional probability of containment bypass decreases by 30% if ISGTR is assumed not to occur.

The licensee has addressed the recommendations of the CPI program.

NOMENCLATURE

AC	Accident Class (or Alternating Current)
AFW	Auxiliary Feedwater
AMSAC	ATWS Mitigating System Actuation Circuitry
AOP	Abnormal Operating Procedure
AOV	Air Operated Valve
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BAST	Boric Acid Supply Tank
BWR	Boiling Water Reactor
CC	Component Cooling System
CCF	Common Cause Failure
CCW	Component Cooling Water
CDF	Core Damage Frequency
CET	Containment Event Tree
CL	Cooling Water
CPI	Containment Performance Improvement
CS	Containment Spray
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DCH	Direct Containment Heating
DHR	Decay Heat Removal
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EP	Emergency Procedure
EPRI	Electric Power Research Institute
FCU	Fan Cooler Units
FW	Feedwater
GKA	Gabor, Kenton & Associates
GL	Generic Letter
GSI	Generic Safety Issue
HEP	Human Error Probability
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
IA	Instrument Air
ICC	Inadequate Core Cooling
IDCOR	Industry Degraded Core Rulemaking
IFL	Internal Flooding
IPE	Individual Plant Evaluation
IPEM	Individual Plant Evaluation Methodology
IPEP	Individual Plant Evaluation Partnership
ISGTR	Induced Steam Generator Tube Rupture
ISLOCA	Interfacing Systems LOCA
LOCA	Loss of Coolant Accident

NOMENCLATURE (Cont'd)

LOCL	Loss of Cooling/Service Water
LOOP	Loss of Offsite Power
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MDAFW	Motor Driven Auxiliary Feedwater
MFW	Main Feedwater
MGL	Multiple Greek Letter
MIC	Microbiologically Induced Corrosion
MOV	Motor Operated Valve
MSIV	Main Stream Isolation Valve
MWe	Megawatt Electric
NRC	Nuclear Regulatory Commission
NSP	Northern States Power Company
PDS	Plant Damage State
PI	Prairie Island
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Analysis
PSF	Performance Shaping Factor
PTS	Pressurized Thermal Shock
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Cooling System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SDC	Shutdown Cooling
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SRO	Senior Reactor Operator
TDAFW	Turbine Driven Auxiliary Feedwater
TER	Technical Evaluation Report
THERP	Technique for Human Error Rate Prediction
TRC	Time Response Correlation
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
USI	Unresolved Safety Issue

1 INTRODUCTION

1.1 Review Process

This technical evaluation report (TER) documents the results of the BNL review of the Prairie Island Nuclear Generating Plant (Unit 1 and Unit 2) 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 on October 4, 1995. Based on this discussion, the NRC staff submitted an RAI to Northern States Power Company on December 21, 1995. The Northern States Power Company responded to the RAI [RAI Responses] in a document dated February 27, 1996. This TER is based on the original submittal and the response to the RAI.

1.2 Plant Characterization

The Prairie Island Nuclear Generating Plant consists of two 560 MWe two-loop Westinghouse pressurized water reactor units, Unit 1 and Unit 2. The reactor coolant system (RCS) of each unit includes the pressure vessel, two vertical steam generators, two reactor coolant pumps, an electrically heated pressurizer and interconnected piping. The RCS is housed inside a large dry containment. Reactors with similar characteristics are: Ginna, Kewaunee and Point Beach 1 and 2.

The plant is located near Red Wing, Minnesota. It is operated by the Northern States Power Co. (NSP). Full commercial operation began on December 16, 1973 for Unit 1 and December 21, 1974 for Unit 2.

A number of design features at Prairie Island 1 and 2 impacts the core damage frequency (CDF). The submittal highlights these features, but does not estimate their effects on the CDF individually.

The following features tend to decrease the CDF:

- The offsite switchyard has a highly reliable and diverse dual ring bus arrangement, minimizing the chance for loss of offsite power. Safeguards buses are normally powered from transformers (1R, 2R or CT11, CT12) which are not required to transfer on loss of the main generator.
- The emergency AC power configuration includes four diesel generators of diverse design and support system requirements. In the event of an SBO condition, each diesel generator has the capability to supply the power requirements for the hot shutdown loads for its associated unit, as well as one train of essential loads of the blacked out unit through the use of manual bus tie breakers interconnecting the 4160 buses between units. The crosstie of a safeguards 4160 bus from one unit to the diesel generator for the other unit can be performed from the control room, within ten minutes of the onset

of SBO. (The present AC power configuration is a consequence of the SBO rule using guidance of NUMARC 87-00 and regulatory guide 1.155. The original AC power configuration included only two DGs shared between units, and the plant was classified as an eight hour coping plant. The plant's present coping classification is four hours.)

As part of the modifications that were performed on the AC power system by adding two DGs after the SBO rule, the safeguards 480 V power system was also modified such that there are now four safeguards 480 V buses per unit where there were only two before. The power supplies for MCCs 1AB1 and 1AB2, as well as 1T1 and 1T2, can be cross tied between Unit 1 and Unit 2. (This crosstie capability, however, was not credited in the plant model.)

- The cooling water system is diverse. It consists of five pumps of which two are horizontal motor driven, and three are vertical pumps with one motor driven and two diesel driven. The single vertical motor driven pump is also backed by safeguard diesel generators (before the SBO rule the pump was powered by a non safeguards power supply) while the two diesel driven pumps do not rely on AC power for operation, i.e., there are three cooling water pumps available following a LOOP. The cooling water system consists of a ring header that can be divided into two separate headers on receipt of an SI signal. Each header supplies half of both trains of safeguards equipment.

- The secondary cooling function involving the main feedwater and auxiliary feedwater systems is very reliable. Main feedwater is lost on a reactor trip but is easily recoverable from the control room. The feedpumps are motor driven, i.e., independent of main steam availability. Each unit has a turbine and a motor driven auxiliary feedwater pump. The motor driven AFW pumps can be crosstied between both units. Large condensate storage tanks (CST) are capable of providing water for decay heat removal without the need for makeup.

- The component cooling systems of the units have crosstie capability. This capability was not credited, however, in the plant model.

- The main steam system design is such that in order to fail the isolation of a ruptured SG, the non-return check valve together with the MSIVs of both the ruptured and intact steam generators must fail which results in a low probability for SG isolation failure.

- The RCP seal cooling is provided by two independent systems, the charging pumps and the component cooling (CC) system. Although the CC system requires cooling water for cooling, the charging pumps do not require CC cooling water to provide RCP seal injection. In addition, the charging pumps do not provide the dual function of SI pumps.

- The injection pressure of the SI pumps is high enough such that operator action is not required to lower the RCS pressure for SI injection following an "S" signal.

- Bleed and feed cooling is a proceduralized action.

- The equipment located in the Auxiliary Building does not require room cooling for extended periods of operation. Analysis has been performed which demonstrates that the CS, CC, RHR and SI pumps do not require ventilation for sustained periods of pump operation.

- The RWST is large and can provide many hours of makeup to the reactor for small break LOCA and SGTR.

- The setpoint for containment spray (CS) actuation is high; 23 psig. This condition preserves RWST inventory for a large fraction of the RCS break spectrum.

- A favorable moderator temperature coefficient for the majority of the cycle allows the plant to effectively ride out an ATWS with feedwater or AFW.

- New cooling water header from corrosion resistant and thicker piping is installed to avoid a potential single flood failing all secondary cooling and instrument air which then causes failure of feed and bleed cooling as the pressurizer PORVs need air to operate.

- The plant does not test any of the valves in the ISLOCA pathways while the plant is above cold shutdown.

- The motor operated valves nearest the RCS in the RHR loop return line and the RHR loop suction line have their power removed during normal operation to prevent inadvertent valve manipulation.

- The low pressure piping in the RHR and SI systems can withstand full RCS pressure without exceeding the piping ultimate pressure stress. (Conditional probability of low pressure piping failure following exposure to full RCS pressure was considered in the ISLOCA analysis.)

The following features tend to increase the CDF:

- Both D1 and D2 diesel generators rely on cooling water for engine cooling functions. (The other two diesel generators are from a different manufacturer and do not require an external cooling medium as they have their own self contained cooling systems.)

- Both main feedwater and feed and bleed cooling are dependent on common support systems; instrument air, cooling water and DC power.

- Cooling water, instrument air and control room chilled water systems are shared by both units. These are systems that are required to be fully operational when either unit is at power. Maintenance on these systems is normally performed while both units are at power. Maintenance activity may influence negatively the availability of these systems.

- The auxiliary feedwater pumps for both units are all located in the same room such that a pipe rupture in the loop A or B cooling water line can result in the failure of all auxiliary feedwater for both units.

- The instrument air compressors are also located in the same room as the auxiliary feedwater pumps such that all the compressors could fail because of a flood, causing loss of instrument air for both units. Loss of instrument air results in closure of the main feedwater regulating and bypass valves which together with auxiliary feedwater failure, results in the loss of secondary cooling. Feed and bleed then fails because the pressurizer PORVs require instrument air to operate.

- Feedwater regulating and bypass valves fail closed also on loss of a train of DC.

- The emergency batteries have only two hours of capacity.

- The instrument air supply to containment has two fail closed air operated valves that are in series on either sides of the containment penetration. Failure of either valve results in loss of instrument air to containment.

- Loss of instrument air will cause the control room chiller outlet cooling water valves to close resulting in loss of chilled water and loss of room cooling to the unit 1 480 V safeguards bus rooms. Without operator intervention, the rooms can heatup and fail the 4160/480 V transformers resulting in a loss of all unit 1 480 V safeguards equipment. This would cause loss of all charging pumps resulting in a loss of all RCP seal cooling causing an RCP seal LOCA in which the SI system would not be available for mitigation.

- Both RCP seal cooling and RCS short term inventory control are dependent on cooling water. Cooling water provides the ultimate heat sink for the component cooling water system which provides cooling to the RCP thermal barrier and the SI pump lube oil coolers. Cooling water also supplies a heat sink for the control room chillers which provide room cooling for the Unit 1 safeguards 480 V bus rooms. On loss of cooling water, room cooling is lost to the Unit 1 safeguards 480 V bus room. Without operator intervention the room can heatup and fail the 4160/480 V transformers resulting in, as previously mentioned, a loss of all Unit 1 480 V safeguards equipment.

  According to the licensee's response to the RAI, since the IPE was submitted room ventilation problems involving the control room chilled water system have changed. Each of the 480 V buses was split into two and now each of the buses carries half of the original loads. In addition, only one of these buses is now located in a room. Furthermore, new thermal analysis showed that without ventilation and assuming no operator action, these room temperatures do not reach debilitating levels within 12 hours. With operator action (to open doors at 12 hours) room temperatures decrease to a steady-state temperature well below operability limits.

- In the licensee's response to the RAI a minor new concern was indicated. The concern is associated with the ventilation of Unit 1 4160 V bus rooms. New bus sequencer units are located now in these rooms and they are sensitive for large temperature change. The risk impact of their failure due to loss of safeguards chilled water was deemed, however, to be not significant, because additional equipment must be lost before the sequencer would be called upon to operate. Even if this were to occur, safeguards equipment could still be operated without sequences (manually) in response to the event. Furthermore, operator actions to restore ventilation to these rooms now are proceduralized.

- The instrument air system has a high failure probability as the system success criterion is such that if two out of three compressors fail, instrument air is considered failed. A single compressor cannot maintain adequate header pressure for both units.

- Feed and bleed cooling is heavily dependent on operator action for success as the operator must manually start an SI pump and open a pressurizer PORV for success.

- Following a small LOCA the SI pumps are the only injection sources that can be used for short term RCS inventory control because the RCS pressure remains above the shutoff head of the RHR pumps. Through plant specific analysis, it was found that depressurizing the SGs to lower RCS pressure to enable the RHR pumps to inject was not possible in time to prevent core damage.

- Given a medium LOCA, switchover to high head recirculation can not be performed from the control room as the RHR to SI crossover motor valves have their breakers locked in the open position. The switchover to high head recirculation must be accomplished within a small time window during which both the SI, CS and/or the RHR pumps are injecting from the RWST. If the operator fails to stop any of the pumps before the RWST level decreases below approximately 5%, all pumps will be damaged as they do not have suction trips.

- The breakers for SI pump suction MOVs from RHR are locked open during power operation.

The Prairie Island Nuclear Generating Station utilizes a large dry containment with a freestanding steel shell construction. The steel reactor containment vessel is enclosed by a 2-1/2 foot thick concrete shield building. Some of the plant characteristics important to the back-end analysis are summarized in Table 1 below and compared to the characteristics of the Zion and Surry plants.

Table 1 Plant and Containment Characteristics for Prairie Island Nuclear Generating Station

Characteristic	Prairie Island	Zion	Surry
Thermal Power, MW(t)	1650	3236	2441
RCS Water Volume, ft^3	5940	12,700	9200
Containment Free volume, ft^3	1,320,000	2,860,000	1,800,000
Mass of Fuel, lbm	108,000	216,000	175,000
Mass of Zircalloy, lbm	24,700	44,500	36,200
Containment Design Pressure, psig	46	47	45
Median Containment Failure Pressure, psig	150	135	126
RCS Water Volume/Power, ft^3/MW(t)	3.6	3.9	3.8
Containment Volume/Power, ft^3/MW(t)	800	884	737
Zr Mass/Containment Volume, lbm/ ft^3	0.019	0.016	0.020
Fuel Mass/Containment Volume, lbm/ ft^3	0.082	0.076	0.097

Both the power level and the containment free volume of Prairie Island are significantly less than those of Zion and Surry. However, the containment volume to thermal power ratio for Prairie Island is between that for Zion and Surry. Similarly, although the reactor coolant system (RCS) water volume, fuel mass, and Zircalloy mass of Prairie Island are smaller than those of Zion and Surry, the ratio of the RCS water volume to reactor thermal power, the ratio of fuel and Zircalloy mass to containment volume for Prairie Island are similar to those for Zion and Surry. The median containment failure pressure obtained in the Prairie Island IPE is 150 psig. This is slightly higher than those obtained in NUREG-1150 for Zion and Surry. 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 upper compartment to the annular containment floors. Flooding of the cavity

is accomplished through two personnel access hatches (located on the instrument tunnel) which are left slightly ajar during normal operation and allows water flow from the containment floor down to the cavity for vessel head flooding. This provides external cooling to the core inside the reactor vessel. Since the flow velocity going into the instrument tunnel may be sufficient to pull the doors closed, a recommendation of securing the hatches open by installing a solid bar or other device, instead of a chain, is discussed in the IPE submittal.

- A steel shell containment that is vulnerable to direct attack by dispersed core debris. The access hatches to the instrument tunnel are in an open area on the basement level of the containment, and for both of the Prairie Island units one of the two hatches faces toward the steel containment, about 30 ft away, with a largely unobstructed path in between. Although a scoping study performed in the IPE shows that the temperature generated by the debris adhering to the steel wall is insufficient to melt the steel and breach the containment, details of the scoping study are not provided in the submittal and the potential effect of corium attack on reducing containment pressure capability is not discussed.

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

- Two separate systems for containment atmosphere cooling and pressure suppression, the Fan Coil Units and the Containment Spray system. According to the IPE, the low conditional probability of containment failure for core damage sequences that do not bypass containment is due in part to the availability of these two completely redundant, diverse means of providing containment heat removal and pressure suppression.

- An Emergency Procedure (EP) that requires the restart of the reactor coolant pumps (RCPs). According to the IPE, the Emergency Procedure "Response to Inadequate Core Cooling" creates the possibility of inducing a steam generator tube rupture (SGTR) during an event in which degraded core cooling conditions already exist. A recommendation of revising the EP is discussed in the IPE submittal.

2 TECHNICAL REVIEW

2.1 Licensee's IPE Process

In order to fulfill the NRC's request concerning Individual Plant Examination the licensee elected to perform a full scope Level 2 PRA. This was documented in the submittal and the subsequent additional information provided by the licensee. The process used to develop this PRA was reviewed by BNL with particular attention to the following areas: completeness and methodology, multi-unit effects and as-built, as- operated status; and licensee participation and peer review.

2.1.1 Completeness and Methodology

The submittal contains the type of information requested by Generic Letter 88-20 and NUREG-1335.

The Industry Degraded Core Rulemaking Individual Plant Evaluation Methodology (IDCOR IPEM) was used in the initial development stage of the Prairie Island IPE. The IDCOR IPEM analysis was completed early in 1991 parallel with the IDCOR efforts, as part of the initial information gathering for the IPE. NUREG-1150 was reviewed, specifically the Surry PRA, since this plant was deemed by the licensee to be the most closely resembling Prairie Island.

The front-end portion of the submitted IPE is a Level 1 PRA. The specific technique used to develop this PRA was the small event tree/large fault tree approach, and it is clearly presented in the submittal.

Internal initiating event and internal flooding were considered. Event trees were developed for all classes of initiating events. The details of the technique applied is extensively described in the submittal: Support systems were modeled in the fault trees of the functional top events and accident sequences were solved by fault tree linking. For deterministic best estimate analysis of reactor and containment response under severe accident sequence conditions, the MAAP 3.0B Revision 19 code was used as principal calculational tool. Mission time for logic model quantification was generally on the order of 24 hours. The consequences of system and equipment failure that might occur during this period were examined well beyond this mission time. Containment response and source term analyses were carried out at least 48 hours to establish important trends in plant response where necessary. Timing and magnitude of potential releases that might occur beyond 48 hours were established based on these trends where necessary.

Sensitivity studies were conducted on initiating event frequencies, operator actions, common cause, test and maintenance and for certain system components. The evaluations were performed to determine the global effect of the parameters of interest. Failure rates were increased/reduced by a factor of 5 where a higher level of uncertainty and variability exists such as human reliability, common cause and test and maintenance unavailability. Failure rates for system components were increased/decreased by a factor of 2 since actual plant data were used and there was less uncertainty associated with these parameters.

The licensee did not evaluate explicitly the changes in CDF due to credited SBO modifications. This was clearly indicated in the RAI responses. (The IPE model would need to be requantified again without the SBO modifications installed.) This review agrees with the licensee's opinion: "Suffice it to say that the SBO modifications did provide a significant reduction in CDF over the plant as it existed before."

In addition to the sensitivity studies, importance analyses were performed on the same quantities. The importance analyses were based on use of the Fussel-Vesely and Birnbaum algorithms. (The results of the analysis are discussed in more detail in Section 2.2.5.3. of this TER.

Sensitivity analyses were conducted also to estimate the reduction in CDF due to certain plant improvements (AFW room fire door closed and procedural changes to allow station air to be cross -tied with instrument air).

The submittal information on the HRA process was minimal. However, the additional information/clarification obtained from the licensee through NRC requests for additional information indicated that the HRA was generally complete in scope.. The HRA process for the Prairie Island IPE considered both pre-initiator actions (performed during maintenance, test, surveillance, etc.) and post-initiator actions (performed as part of the response to an accident). While the analysis of pre-initiator actions considered both miscalibrations and restoration faults, only restoration faults were explicitly modeled. The potential for miscalibration of a group of sensors or instruments was assumed to be included in the common cause failure modeling of the instruments themselves. With the exception of events excluded on the basis of a qualitative screening applied during the pre-initiator human action selection process, all pre-initiator restoration errors were quantified. One of two HEPs (0.003 or 0.01) was assigned to each event depending on whether or not a post-maintenance verification of component status was required by procedure. Plant-specific component unavailabilities were then calculated using the HEPs in conjunction with factors such as maintenance frequency and duration, test frequency and interval, refueling outage frequency, and time from completion of the corrective maintenance to the retest of the component.

Post-initiator human actions modeled included both response-type and recovery-type actions. The submittal indicates that screening values were assigned "by following a flow chart and answering a series of questions." Neither the questions nor the flow chart was provided, but "plant-specific estimates of the time available to initiate and perform the action" and "PSFs associated with degree of difficulty and stress" were considered. The screening HEPs were based on the Handbook (NUREG/CR-1278), Wash-1400 (NUREG-75/014), and "the data sources used in the IDCOR BWR IPE methodology." Post-initiator response type actions found to be important received detailed HEP development. An action was important if it contributed significantly to baseline core damage probability or "if a change in the failure rate could cause significant increase in overall core damage probability." Most of the actions receiving detailed analysis were quantified with the ASEP method (NUREG/CR-4772). The response to the NRC RAI states that a more refined estimate for the five most important human actions was obtained using the Handbook (NURG/CR-1278). The licensee's response to the RAI also indicated that most recovery probabilities were derived from NSAC-161, "Faulted Systems Recovery Experience" and that recovery actions were only added to cutsets when it was apparent that an operator would have sufficient time to perform the additional action. Local recovery of valves which failed to open or close was credited only when there was control room indication of valve position and the valve was easily accessible . A non-recovery probability of 0.025 was assigned to these events. Recovery of equipment such as pumps had the same criteria, but the non-recovery probability was "approximately 0.5." Plant-specific performance shaping factors and dependencies (such as those among multiple actions in a sequence) were apparently considered for both response and (most) recovery actions. Human errors were identified as important contributors in accident sequences leading to core damage and several recommendations to improve procedures and operator training were provided.

The methodology employed in the Prairie Island IPE submittal for the back-end evaluation is clearly described. Containment event trees (CETs) were developed to determine the containment response and ultimately the type of release mode given that a core damage accident has occurred. The front-to-back end interface are provided in the IPE by the definition of 14 Accident classes (ACs). These accident classes are identified by a three-character designator addressing the accident initiator, core melt timing, and the RCS pressure at the time of core melt. Unlike the PDSs used in some other IPEs, the availability of containment systems are not explicitly included in the definition of the ACs. In the Prairie Island IPE, containment system fault trees (for containment spray injection, containment spray recirculation, and containment fan coil units) are quantified as frontline systems, along with the Level 1 frontline and support system fault trees, using linked fault tree models. The containment systems fault tree cutsets are input to the CET branches as necessary to support CET quantification.

The CETs used in the Prairie Island IPE provide a structure for the evaluation of all of the containment failure modes discussed in NUREG-1335. The containment failure modes that are assumed negligible and thus not included in CET quantification include those from melt-through of the containment steel shell, vessel thrust force (the rocket mode failure), and penetration failure due to degradation of sealing materials under harsh environmental condition. The containment failure modes that are considered as unlikely but are assigned small probability values include those from direct containment heating, in-vessel steam explosion, ex-vessel steam explosion, and hydrogen combustion. Containment isolation failure is also considered as unlikely but is evaluated in CET quantification using the data obtained in containment isolation analysis.

The quantification of the CET in the Prairie Island IPE is based on plant-specific phenomenological evaluations. The evaluations include modeling and bounding calculations (based upon experimental data), consideration of phenomenological uncertainties, and MAAP calculations. The result of the CET analysis are grouped to seventeen CET end states. Release fractions for these CET end states are determined by the analyses of representative sequences using MAAP computer codes.

To complete the IPE process, the comparisons were made with the results of other PRA studies performed for similar plants, like Kewaunee and Point Beach. The comparison showed, that several of the Prairie Island (PI) accident sequences have similar frequencies as those obtained at these 2-loop Westinghouse plants of the same vintage. The comparison identified also, however, that some aspects of the Prairie Island plant design lead to different results than those obtained in the PRAs performed for these plants. (Section 2.4.2 of the submittal.)

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

To develop the IPE, a wide variety of up-to-date information sources were used: the Updated Final Safety Analysis Report (UFSAR), current Technical Specifications, Plant Operations Manuals, Emergency Operating Procedures, Plant Surveillance Procedures and Plant Computer Files, Work Requests, plant drawings, vendor technical manuals, design basis documents etc. Plant walkdowns were performed for fault tree constructions and spatial interactions for internal flooding effects.

The IPE was performed for the plant design as it existed in the fall of 1993. The relatively recent plant configuration includes changes made to the AC power distribution system that are important to accident sequences associated with loss of offsite power initiators and station blackout.

The plant model considers dual unit effects, those that are explicitly taken into account in the Prairie Island plant model belong to three categories:

1. Shared systems or systems capable of being crosstied between units,
2. Dual Unit Initiators, and
3. Common Cause modeling.

Shared Systems: Instrument air, Cooling Water, and Control Room chilled water systems.

Cross Tied Systems: Emergency AC power, and Auxiliary Feedwaters systems. The component cooling water system is also capable being crosstied between units. It was not modeled this way, however, due to "the low contribution to the overall CDF from CC" (cause given in the submittal),and to reduce modeling effort (cause mentioned in RAI response). The IPE does not take credit for this feature of the system.

Dual Unit Initiators (marked by asterisk in Table 2 "The PI Initiating Events"): For these events, equipment which could be crosstied from Unit 2 was not credited until it was demonstrated that the equipment was not required for a potential Unit 2 transient or could support both units simultaneously.

Common Cause Modeling: Analyses for similar components in systems that are shared or could be crosstied was performed by considering the coupled systems as a whole.

Procedure reviews, discussions with control room personnel, and walkdowns of selected local operator actions and control room panels helped assure that the IPE HRA represented the as-built, as-operated plant. The licensee conducted an extensive data collection effort to develop plant specific initiating event frequencies and component failure rates. The initiating event frequencies were derived from data collected for an eleven year period between 1980 and 1990. Plant specific component and maintenance unavailability data were collected for a ten year period between 1978 and 1987. These data were used in both Level 1 and Level 2 event trees and fault trees.

A review of the operator actions identified several that related to multi-unit effects. In fact, two of the most important human actions involved multi-unit interactions. One action was to cross tie unit 2 AC power to unit 1 during a station blackout of unit 1. Another involved a cross tie of the unit 2 motor driven AFW pump to unit 1 to restore MFW. Several versions of this event were quantified to reflect different contexts, e.g., presence or absence of an "S" signal. Procedure reviews, plant walkdowns, and discussions with operations and training staff helped assure that the IPE HRA represented the as-built, as-operated plant. In addition, the licensee stated that no recommended improvements were credited in the reported IPE CDF. However, in the response to the RAI, it is indicated that credit was taken in the level 2 analysis for two procedure changes. One involved changing an emergency procedure to direct the operators to check for adequate steam generator level before attempting to start an RCP. The intent was to minimize the potential for induced SGTR. The second recommendation credited was to make plant design or administrative control changes to ensure that the sump C hatch doors remain open to allow water injected from the RWST following an accident to flow through the doors to the reactor cavity.

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

The submittal explicitly indicates (see, e.g., the submittal's cover letter, dated March 1, 1994) that the licensee intends to maintain the IPE as a "living" PRA. This is to provide continuing insights for design change review, procedure enhancements, training, and shutdown risk planning.

2.1.3 Licensee Participation and Peer Review

Licensee participation in the IPE process and review activities are discussed briefly in Section 1 of Part 5 of the IPE submittal. The group responsible for all PRA-related activities at Prairie Island is the Self Managed Work Team of the Northern States Power Company (NSP), headed by the Director of Licensing and Management Issues. The NSP PRA staff was made up of five engineers and one engineer associate. Two people worked at the Prairie Island site and the rest at the General Office. This personnel was involved in all aspects of the analysis. In-plant expertise was ensured by the fact that two individuals maintained SRO certification at the plant, one of them previously held an SRO license. There was also experience in other nuclear areas, such as: core transient analysis, operations, system engineering, plant technical staff, nuclear Navy and reactor physics. TENERA, Westinghouse Electric Corporation and Fauske & Associates Inc., which are part of IPEP (Individual Plant Evaluation Partnership) contributed to help NSP developing the PRA/IPE. Gabor Kenton & Associates (GKA) also provided consulting services.

The NSP PRA/IPE team has been very active in applying PRA methodology to ensure plant safety. Table 5-1 of the submittal lists a significant number of these activities.

The IPE reviews included both independent in-house reviews and an external review. The first independent in-house review involved the verification of the calculations, i.e., the assumptions used and the correctness of the results. This review was done by someone other than the preparer. The second review was a review of other PRA-analyses performed in the industry (i.e., IPE reports submitted to NRC by Kewaunee and Point Beach, Surry NUREG-1150 and NUREG-4550). The third review was performed by the Senior Review Team. This team consisted of four industry experts, which reviewed the PRA/IPE to ensure correctness of the methodology and that the results are consistent with other PRA's in the industry. The fourth review was an independent in-house review done by NSP personnel that was not involved in the development of the PRA.

The RAI response details this review: "The draft submittal was distributed widely to the plant staff for review and comment. The reviewers were asked to pay particular attention to portions of the report that fell within their area of expertise. Many comments and questions were raised during the review process. These were studied and changes were made to the model and/or the report as necessary to answer the reviewer's questions and comments."

Neither the submittal nor the RAI response discuss the nature and the quality of the review questions and their resolutions.

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 IPE initiating event analysis used the work performed previously in the IDCOR IPEM, but significantly enlarged its scope and employed updated data. Table 2 presents the list of initiating events and provides the frequency for each initiating event. The column "SOURCE" of the Table identifies the origin and method of the frequency value indicated in the Table; whether it was derived from operating experience or generic data, or whether it was determined from plant-specific analysis using NRC and/or industry documents.

The total number of initiating events is 23. They are grouped into seven categories (the internal flooding events and anticipated transients are classified together and the different ATWS events are collapsed into one category), since it was found that the minimal number of significantly different ways in which the plant responds to challenges is seven.

Transient occurrence data (from the period 1/1/80-12/31/90) were used to derive the plant-specific initiator frequencies (simply by dividing the number of events by the number of years of data). Generic initiating event frequencies were obtained from the published sources given in the table. The SGTR initiator frequency was obtained by Bayesian update. Plant-specific system fault tree models were used to estimate the special transient initiating event frequencies (loss of DC Train A and B, loss of Instrument Air, etc.) and Interfacing System LOCAs. The main feedwater and main steam line breaks were quantified by performing a review of piping from the steam generator to the outside isolation valve (MSIV, feedwater regulating and bypass valves). A generic pipe rupture frequency was then applied to these piping sections. Failure of the isolation function was handled in the main steam line/feedline event trees.

In the RAI response the licensee provided reasons why certain initiators were omitted from the plant model and explanations about the approach used in the determination of certain initiating event frequencies. Brief summaries are given below:

a) Loss of 120 V vital AC bus events were not considered as initiating events as loss of a single 120 V AC bus will not cause an automatic or imminent reactor trip. Loss of a 120 V bus is also permitted by Technical Specifications. to occur for a period of 6 hours before actions need to be taken to shutdown the affected unit. Loss of two or more 120 V AC bus events, while they cause reactor trip, were also neglected as initiating events because they are very rare: "this type of failure has not occurred over the operating history of the plant and there are no published data sources with dependable failure rates for such initiator."

b) Common cause failure of DC buses was not modeled as an initiating event due to low frequency.

c) Common cause failure of the safeguards 4160 V AC buses was not modeled as an initiating event because of the low frequency of the common cause failures of the buses, and because such a common cause failure does not results in a plant trip, only a forced manual shutdown.

d) Loss of the safeguards control room chilled water system was not included in the IPE special initiating event analysis, because such an event causes no direct plant shutdown or trip and unavailability of safeguard equipment without assuming additional equipment or operator action failures. The licensee justified the exclusion of this initiator (by new room heat-up analyses and by the extensive changes in the electrical and other systems served by this chilled water system) since the IPE was submitted. The licensee considers that the CDF contribution of this initiator is insignificant and no meaningful information would have been gained if this initiator had been modeled in the submittal.

e) The Prairie Island Reactor Pressure Vessel is thought to be not as susceptible to pressurized thermal shock (PTS) due to the low percentage of copper in the RPV weld joints. Inspection of irradiated RPV samples has also confirmed that the vessel is aging slower than expected from the effects of neutrons and radiation. The reference temperature for PTS is 208°F at the most limiting reactor vessel weld. Plants with higher reference temperature have an estimated CDF due to PTS of about 1E-08/yr, which is considered to be a negligible contribution to CDF at Prairie Island. With the exception of PTS, no specific credible mechanism for RPV failure has been identified and therefore RPV rupture was not explicitly included as an initiator.

f) The IPE model apparently neglects the small-small LOCA initiating events. According to the licensee, this initiator is implicitly included in the model: partly in the small LOCA initiating events (breaks larger than 3/8"), and partly in the normal transients (breaks below 3/8").

g) The steam line break and feedwater line break initiating event calculation considered piping only within the containment for two reasons: 1. The frequency of any non isolated line break beyond the first isolation valve is considered to be insignificant (the low probability of pipe rupture combined with the high probability of isolation valve closure) with respect to plant risk; 2. A break outside containment, would be contained by the auxiliary building or turbine building steam exclusion boundary. This boundary consists of walls, doors and other barriers to prevent the propagation of a steam environment to safeguards equipment. Openings in these boundaries are under tight administrative control. In addition, required safeguards equipment within the steam environment is qualified for harsh environment.

h) The licensee states that the frequencies for Loss of Service Water and Loss of Component Cooling Water initiating events, which were calculated by fault trees, imply pipe rupture probabilities; they were added to the quantified fault tree results.

It should be noted that this plant did experience an SGTR event due to a loose part in the RCS post-maintenance, which is reflected in a somewhat higher SGTR frequency.

The present review finds the set of initiating events selected for IPE analysis to be acceptable and the initiating event frequencies to be reasonable and comparable to other PRA studies.

2.2.1.2 Event Trees

The IPE developed eight event trees to model the plant responses to internal initiating events: large, medium and small LOCA event trees, SGTR event tree, main steam line/feedwater line break event tree, transient event tree, loss of offsite power (including station blackout) event tree and ATWS event tree.

- No event trees were developed for interfacing systems LOCAs but each ISLOCA source was separately analyzed. The most likely sources identified were: RHR cold leg injection, RHR low head SI to the reactor vessel and RHR suction from the hot legs. No credit was given for the operator to locally isolate an ISLOCA pathway. Core damage was assumed on the rupture of any of these piping systems outside containment. It was also assumed, that the low pressure piping mainly will break in the CS pump room causing the failure of the CS, SI and RHR pumps resulting in core damage due to loss of short term RCS inventory. If the low pressure RHR piping did not instantaneously rupture, it was assumed, the RHR pump seals would fail causing loss of both RHR pumps. In this case, operator action to cooldown and depressurize the RCS to minimize the flow out the RHR pump seals and preserve RWST inventory was credited.

- No event tree was developed for the reactor vessel rupture event as it is assumed to lead directly to core damage.

- No separate event trees were developed for flooding scenarios, the transient event tree was used with additional flood-caused failures flagged in the appropriate fault trees.

The event trees developed for the Prairie Island IPE are small and structured around safety functions.

The event tree end states are divided into two possible outcomes: success or core damage. The core damage end states then are classified into 14 accident classes.

- Core uncovery and core damage were assumed to occur when core exit thermocouple temperature exceeds 1200°F for 30 minutes or whenever they reached 2000°F. Timing associated with this condition was obtained through MAAP analysis of sequences in the various accident classes. Typical time frames for this condition were found to range from 2 hours for transients following loss of secondary cooling to 30 minutes for large LOCA without RHR injection.

  The RAI response extensively discusses these core damage criteria, which is one of the fundamental aspects of the IPE analysis. From the discussion only one portion is reproduced here: "Previous PRA studies (NUREG-1150 and 4550) conservatively assumed core uncovery signified core damage. This would assure that there would be no release of fission products from the fuel to the RCS. Therefore the maximum release of the fission products from the RCS would be limited to the initial inventory in the RCS prior to the initiating event. Subsequent studies have shown that steam cooling on the upper portion of the core (after core uncovery) can be effective in delaying or preventing core damage."

- The success criteria of safety systems associated with the safety functions (the top events in the event trees) were derived from plant specific analyses of system responses to transient and LOCA initiating events. According to the submittal, the basis for the success criteria was a combination of realistic calculations using MAAP, USAR and operations manual descriptions. If a system served more than one function, it's success criteria varied for each function.

Like some other PWR IPEs, the Prairie Island IPE assumes (calculates) also that core flood tanks are not needed in large and medium LOCAs. In regard to this an RAI response notes the following:"By assuming accumulators were necessary for LOCA sequences, the resulting accident sequence probabilities will not be impacted as accumulators are passive components at Prairie Island. The probability for the

failure of both accumulators is approximately 1E-06, which when multiplied by the LOCA frequencies becomes an insignificant contributor to CDF."

The success criteria for RCP seal cooling is associated with the operation of component cooling to the thermal barrier heat exchangers or operation of a charging pump supplying seal injection to the seals. The IPE uses the Westinghouse RCP seal LOCA model that models core uncovery due to seal failure as a function of time from loss of seal cooling and includes the effects of restoration of offsite power. Two cases were considered: one with RCS cooldown and one without. The model represents unqualified Westinghouse RCP seal O-ring material as this is what is installed in the RCP seals at Prairie Island. The RAI response provides some more details about the RCP seal LOCA analysis and points out that the tripping of the RCPs by an operator, given a loss of component cooling is not explicitly modeled in the IPE, but the loss of CC is. The RAI response provides also an estimate of the total contribution of the RCP seal LOCA to the total CDF. It is 9.5E-06/yr, which is approximately 19% of overall CDF.

2.2.1.3 Systems Analysis

A total of 21 systems/functions are described in Section 3.2.1 of the Submittal. Included are descriptions of the following systems: reactor protection, AMSAC, CVCS, safety injection, RHR, main feedwater and condensate, auxiliary feedwater, pressurizer PORV, containment fan coil units, containment spray system, cooling water, component cooling water, instrument air, onsite AC power, DC power, 120 V instrument AC power, SI signal, safeguards chilled water, SG PORV, MSIV, room cooling.

Each system description includes a discussion of the system design and operation, dependencies, success criteria and role in safety function(s).

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

2.2.1.4 System Dependencies

The following types of dependencies are considered in the IPE:

a. Safety function (front line system) dependency on the initiating events (Table 3.2-3 of the submittal),
b. Front line system dependency on other frontline systems (Table 3.2-4 of the submittal),
c. Front line system dependency on support systems (Table 3.2-5 of the submittal),
d. Support system dependency on other support systems. This is provided in Table 1 of the plant's response to the RAI, because it was inadvertently left out of the IPE submittal.

These dependencies are presented in the IPE documents in matrix form. The matrices have notes providing detailed information on: shared components, instrumentation and control, isolation, motive power, direct equipment cooling, areas requiring chilled water for room cooling and operator actions. Room cooling by chilled water was determined to be important (more in the submittal and somewhat less in the RAI response) in the 4160 V/480 V bus rooms, including event monitoring rooms, switchgear room, relay room, computer room, control room air handler room, and in the RHR pits. In other rooms, temperature limits would not be exceeded in the accident scenarios of interest.

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 with fault tree linking to quantify core damage sequences. Fault tree models were developed for the functional top events depicted in the event trees. These high level fault trees, however are not shown, only explained qualitatively ("defined") in the submittal. The front line systems in these high level fault trees were also modeled by fault trees, as were their support systems. The submittal does not show them either. The EPRI's CAFTA fault tree manager was used for development and quantification of the top event fault trees. They were linked together using Logic Analyst's HPSETS code to determine the accident frequencies. The front line systems fault trees were developed to allow the support system fault trees to be linked directly into the logic when quantification was performed. Human errors were included in the fault trees, where an operator action was necessary in order for a system to operate, such as the operator to switchover to recirculation. Operator actions in response to equipment failures were not included in the fault trees, but were later included after sequence quantification as recovery factors.

The cut set truncation limit used was 1.E-09. In the fault trees a component was not modeled further down if no useful insights could be gained by more detailed modeling, i.e., if all the failures of the subcomponents were encompassed by one failure mode of interest, such as, pump fails to start. Similarly, negligible faults associated mainly with passive components such as pipes and manual valves were eliminated from further consideration.

- The fault trees developed for the Prairie Island IPE are listed in Table 3.

Table 2 Initiating Events

No.	Category Name	Initiating Event	Designator	Frequency (per yr)	Source[a]
1	LOCA's	Small LOCA	SLOCA	3.00E-03	PWR IPEM Methodology
		Medium LOCA	MLOCA	8.00E-04	
		Large LOCA	LLOCA	3.00E-04	
2	Anticipated Transients	RX Trip (other than below)	TR1	1.68	Plant Data
		SG HI-HI LVL	TR2	9.00E-02	Plant Data
		Inadvertent SI-Signal	TR3	2.30E-01	Plant Data
		Loss of Feedwater	TR4	9.00E-02	Plant Data
	Internal Flooding	Aux. Bldg. Zone 7 (695' El)	AB7FLD	5.05E-03	EPRI TR-102266
		Aux. Bldg. Zone 8 (above 695')	AB8FLD	1.34E-04	EPRI TR-102266
		TB Bldg. Zone 1 (AFWP Rm)*	T1FLD	1.04E-05	EPRI TR-102266
		TB Bldg. Zone 13 (Relay Rm)*	T13FLD	2.68E-05	EPRI TR-102266
		Scrnhse Zone 1 (SG Cl Area)	SH1FLD	6.09E-06	EPRI TR-102266
		Scrnhse Zone 2 (Non-SG Area)	SH2FLD	2.54E-03	EPRI TR-102266
3	Special Transients	Loss of Cooling Water*	LOCL	1.82E-05	Fault Tree
		Loss of Comp. Cool Water	LOCC	3.46E-03	Fault Tree
		Loss of Train A DC Power	LODCA	8.69E-03	Fault Tree
		Loss of Train B DC Power	LODCB	8.69E-03	Fault Tree
		Loss of Instrument Air*	INSTAIR	1.17E-02	Fault Tree
4	Unanticipated Transients	Main Feedwater Line Break	MFLB	2.50E-05	WASH-1400
		Main Steam Line Break	MSLB	3.90E-04	WASH-1400

Table 2 Initiating Events (Cont'd)

No.	Category Name	Initiating Event	Designator	Frequency (per yr)	Source[2]
5	Loss of Offsite Power	Loss of Offsite Power*	LOOP	6.50E-02	NUREG-1032 NUMARC-8700
6	LOCA Outside Containment	Interrystem LOCA	ISLOCA	2.27E-07	Fault Tree, NUREG-5102
		Steam Generator Tube Rupture	SGTR	1.50E-02	Plant Data
7	Failure to Trip (ATWS)	RX Trip (Other than Below)	ATWS-TR1	2.52E-05[1]	
		SG HI-HI Level	ATWS-TR2	1.35E-06[1]	
		Inadvertent SI Signal	ATWS-TR3	3.45E-06[1]	
		Loss of Feedwater	ATWS-TR4	1.35E-06[1]	
		Loss of Cooling Water	ATWS-LOCL	2.73E-10[1]	
		Loss of Comp. Cooling Water	ATWS-LOCC	5.19E-08[1]	
		Loss of Train A DC Power	ATWS-LODCA	1.30E-07[1]	
		Loss of Train B DC Power	ATWS-LODCB	1.30E-07[1]	
		Loss of Instrument Air	ATWS-INSTAIR	1.76E-07[1]	
		Loss of Offsite Power	ATWS-LOOP	9.75E-07[1]	
		Main Feedwater Line Break	ATWS-MFLB	3.75E-10[1]	
		Main Steam Line Break	ATWS-MSLB	5.89E-09[1]	
		Int. Fld. AB Zone 7 (6951) El)	ATWS-AB7	7.58E-08[1]	
		Int. Fld. AB Zone 8 (Above 695)	ATWS-AB8	2.01E-09[1]	
		Int. Fld. TB Zone 1 (AFWP Rm)	ATWS-TB1	2.55E-10[1]	
		Int. Fld. TB Zone 13 (Rly Rm)	ATWS-T13	4.02E-10[1]	
		Int. Fld. SH Zone 1 (SG Area)	ATWS-SH1	9.14E-11[1]	
		Int. Fld. SH Zone 2 (Non-SG)	ATWS-SH2	3.81E-08[1]	
		Small LOCA	ATWS-SLOCA	4.50E-08[1]	
		Medium LOCA	ATWS-MLOCA	1.20E-08[1]	
		Steam Generator Tube Rupture	ATWS-SGTR	4.50E-07[1]	

Initiating Events Notes:

* Dual Unit Initiators:—

(1) All frequencies are multiplied by the failure to trip demand rate (1.5E-05/d) to determine ATWS frequency.

(2) Sources: "Plant Data" indicates plant-specific operating experience data. "Fault Tree" indicates that a plant-specific fault tree was constructed and quantified for this system to determine the initiating event frequency. Reference to an NRC or industry document indicates that a plant-specific analysis was performed to determine the initiating event frequency using the methodology provided in that document.

Table 3 Fault Trees Developed in the Prairie Island IPE

Safety Functions/Frontline Systems	Support Systems
Reactivity Control Chemical Volume and Control System	Station Power Emergency AC Power DC Power
Secondary Heat Removal Auxiliary Feedwater Main Feedwater/Condensate Steam Generator PORVs	Cooling Water Component Cooling Water Instrument Air SI Actuation Signal Instrument Power
Short Term Injection Safety Injection Pressurizer PORVs Low Head Injection (RHR)	CS Actuation Signal
Long Term Injection High Head Recirculation (SI/RHR) Low Head Recirculation (RHR)	
Containment Control Containment Spray Containment Spray Recirculation Fan Coil Units	
RCS Cooldown and Depressurization Charging RHR Shutdown Cooling Auxiliary Spray	

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. Formal mathematical uncertainty analysis was not performed on the results. However, the submittal reports point estimates of the core damage frequencies by initiating events separately for Unit 1 and Unit 2. In addition it provides point estimates for the important sequence frequencies of Unit 1.

Importance analyses (by using Fussel-Vesely and Birnbaum importance indicators) were performed for the initiating events, the major operator actions and all the systems considered in the plant risk model. The importance analysis also included the corrective maintenance contributions as well as the preventive maintenances and test contributions to the overall CDF.

Sensitivity analyses were conducted on initiating event frequencies, operator action failure rates, common cause, test and maintenance and for certain system components (e.g., EDG failure rates).

In response to an RAI, an NSP review of the modeling of maintenance unavailabilities identified several cases in which maintenance on opposite unit equipment for shared and cross-tied systems was not correctly accounted for in the IPE. The combined impact of these omissions was then estimated by incorporating them into the IPE model and quantifying the change in CDF. The RAI response does not specify the affected sequences individually; it provides only the approximate value of the total CDF rise, which is 6% (i.e., the total CDF is approximately 5.3 E-05/yr). It remarks, however, that nearly all of this increase is due to the additional (preventive maintenance) unavailability of Bus 25, which supplies power to the 21 AFW pump.

The RAI response offers also some explanation about the modeling omission of pressurizer PORV block valves. In the IPE the PORVs are included in the model for feed & bleed and ATWS. However, in neither case are they modeled such that the block valves are required to open.

In the ATWS modeling, the fraction of time that either block valve was assumed to be closed due to leaking PORVs was estimated as one month per year. That estimate results in an unavailability value of 8.3E-02 for each block valve. The impact of this omission to the ATWS CDF is not estimated.

In the case of the feed & bleed, however, the omission increases only the feed & bleed unavailability by an amount on order of 1E-06, much smaller than the unavailability due to human error, which is 4.5E-02, thus its effect on the CDF is negligible.

2.2.2.3 Use of Plant Specific Data

- The submittal emphasizes that many initiating events and all major and significant mechanical component failure rates were generated using plant specific data. For component failure data collection the sample period of 1/1/1978-12/31/1987 was selected, because this time period was readily accessible using electronic data retrieval techniques.

  During the component failure data collection process system boundaries were established which were used also for system fault tree modeling (e.g., the EDG boundaries included the engine with it's support systems, the governor, the generator, the output breaker, and all control circuitry). Both demand and time related component failure rates were determined.

  The licensee states that the plant specific failure rates were compared to generic sources to check for reasonableness. When a discrepancy occurred the plant specific value was thoroughly checked. Whenever sufficient plant data were available to make a statistically acceptable estimate of the failure rate, the plant value was used. However, for some components and failure modes which had no recorded data at the plant, generic values were applied.

  Plant specific data were applied also to derive the fraction of time a given component or train of equipment could be expected to be out of service for maintenance. Information to determine component unavailability during testing was obtained from a review of plant surveillance procedures.

- The submittal presents both the generic failure rates (used for certain components) along with the list of equipment for which plant specific data were used (Tables 3.3-1 and 3.3-2 of the submittal).

  In this review Table 4 compares the IPE's plant specific failure data for selected components with values typically used in PRA and IPE studies, using the NUREG/CR-4550 data as a basis for

comparison [NUREG/CR 4550, Methodology]. One can see, that generally the Prairie Island data tend to be smaller than NUREG/CR-4550 data, except for the DG and AFW pump "fail to run" failure rates, and for the indicated MOV, AOV failure rates. In some cases, for example for IA compressors and check valves, the differences reach more than an order of magnitude.

The cause of these differences is apparently "statistical fluctuation", or "fact of life". The RAI response depicts for instance the data analysis of check valves, as follows:

> "To calculate failure rates for valves all valves or check valves were pooled together to create a large data pool of demands and failures. For check valves this created a pool with over 25,000 open demands and an equal number of close demands. There were no failures of check valves found in the plant records. However, since no equipment can be assumed to have a zero failure probability, a value of 0.5 was used for the number of failures. The resulting failure rate is lower than generic industry values, however, the large pool of demands coupled with the lack of failures justifies using the plant specific value."

2.2.2.4 Use of Generic Data

The major components for which generic failure data were used in the IPE are the following: Pressurizer PORV, Circuit Breaker (4160 V/480 V), solid state logic modules, inverters, battery and battery chargers, MCCs, heat exchangers and chillers, relief valves, fans, relays, switches and transmitters. Generic values were used also for some failure modes on some components due to low demands or operating hours which did not produce reliable plant specific failure values (e.g., for the "fail to run" failure mode of the SI/CS pumps).

Common cause failures of circuit breakers, switchgear and relays were not explicitly modeled. According to the RAI response they were considered implicitly. For instance, common cause failures of loads supplied through the breakers, such as pumps, valves and other components that can be attributable to common cause mechanisms, were modeled. Common switchgear failure was implicitly analyzed (in terms of function and the effects of failures) with other failures, such as diesel generator common cause failures. Relay common cause failures were considered to be covered under common cause failures of instrumentation and control trains.

The following documents were used as "sources " for the generic data :

1. NUREG/CR-2815, Probabilistic Safety Analysis Procedures Guide,
2. NUREG/CR-4550, Analysis of Core Damage Frequency,
3. IEEE Standard 500 Data,
4. EPRI TR-100320, Vol. 2, Reliability Centered Maintenance (RCM) Technical Manual,
5. WASH-1400, Reactor Safety Study (NUREG 75/014).

2.2.2.5 Common-Cause Quantification

The Prairie Island IPE team systematically examined the plant model for redundant components to address potential common-cause failures (within individual systems and across both units). The component groups for which common cause events were defined are given below:

1. Diesel generators (failure to start and run),
2. Pumps (failure to start and run),
3. MOVs and AOVs (failure to open and close),
4. PORVs (failure to open or reclose on demand),
5. Check Valves (failure to open on demand; failure to reclose),
6. Batteries (failure to operate on demand),
7. Instrumentation and Control components (failure to send signal or actuate equipment),
8. Air compressors (failure to start and run),
9. Cooling fans (failure to start and run),
10. Chillers (failure to start and run).

The common cause probability model used was the Multiple Greek Letter (MGL) method.

The primary data for the common cause factor estimates were taken from the documents given below:

NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," EPRI, June 1985, NUREG/CR-3289, "Common Cause Fault Rates for Instrumentation and Control Assemblies," U. S. NRC, May, 1983, NUREG/CR-2770, "Common Cause Fault Rates for Valves," U. S. NRC, February, 1983, "Nuclear Plant Reliability Data System," INPO.

These data were then sorted and classified according to the usual procedures for treating common cause failures in safety and reliability studies. The obtained β and (if applicable) γ and δ factors are reported in the submittal, with discrimination based on the failure modes (e.g., in general, different values of MGL parameters are given to failure to start as opposed to failure to run).

For convenience, the β factors for various combinations of component failures are reproduced in Table 5 of this report to compare them with some reference β factors suggested in NUREG/CR-4550. The reference β factors are also presented in Table 5 (NUREG/CR-4550 reports only failure to start β factors).

Based on the table's data, the general observation is, that the common cause parameters applied in the Prairie Island IPE seem reasonable and consistent with those of recommended in NUREG/CR-4550 (in addition, the common cause factors for the turbine drivers of the AFW pumps seem also reasonable: for FTS = .17 and for FTR = 0.04).

2.2.3 Interface Issues

2.2.3.1 Front-End and Back-End Interfaces

The submittal treats this issue rather concisely. It states that similar core damage sequences were grouped into classes according to the following criteria:

- Integrity of the containment
- Initiator type
- Relative timing of the core melt with respect to the initiator
- Primary system pressure.

Table 4 Comparison of Component Failure Data

Component	Prairie Island+		4550*
MDAFW Pumps	*Pump Motor*	*Pump*	
fail to start	7.6E-04	6.1E-04	3.0E-03
fail to run	1.4E-04	1.3E-04	3.0E-05
TDAFW Pumps	*Turbine*	*Pump*	
fail to start	9.4E-03	6.1E-04	3.0E-02
fail to run	6.1E-03	1.3E-04	5.0E-03
SI Pumps			
fail to start	1.1E-03		3.0E-03
fail to run	3.0E-05*		3.0E-05
RHR Pumps			
fail to start	4.6E-04		3.0E-03
fail to run	2.6E-05		3.0E-05
11/21 CL (SWS) Pumps			
fail to start	4.8E-04		3.0E-03
fail to run	1.2E-05		3.0E-05
CCW Pumps			
fail to start	2.7E-04		3.0E-03
fail to run	2.7E-06		3.0E-05
IAI Compressors			
fail to start	2.0E-04		8.0E-02
fail to run	6.9E-06		2.0E-04
Check Valves			
fail to open	1.8E-05		1.0E-04
fail to close	1.8E-05		1.0E-03
MOV			
fail to open	4.7E-03		3.0E-03
fail to close	7.6E-03		
Air Operated Valve			
fail to open	2.6E-03		
fail to close	1.7E-03		2.0E-03
Diesel Generator (D1 and D2)			
fails to start	3.4E-03		3.0E-02
fails to run	1.1E-02		2.0E-03

+Values indicated are rounded for comparison.

*Generic Failure Rate from NUREG/CR-4550

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.

Table 5 Comparison of Common-Cause Failure Factors

Component	Failure Mode	Submittal β Factor	Reference β Factor
FW pump only, CCF of 2 pumps	FTS/FTR	0.035/0.085	0.056
AFW motor driver, CCF of 2 drivers	FTS/FTR	0.15/0.013	
RHR pump, CCF of 2 pumps	FTS/FTR	0.16/0.18	0.15
SI pump, CCF of 2 pumps	FTS/FTR	0.16/0.17	0.21
CL pump, CCF of 2 pumps	FTS/FTR	0.018/0.084	0.026
CL pump, CCF of 3 pumps	FTS/FTR	0.018/0.099	0.014
CCW pump, CCF of 2 pumps	FTS/FTR	0.14/0.058	0.026
CS (spray) pump, CCF of 2 pumps	FTS/FTR	0.38/0.081	0.11
MOV, CCF of 2 valves	FTO/FTC	0.078	0.088
MOV, CCF of 3 valves	FTO/FTC	0.082	0.057
AOV, CCF of 2 valves	FTO/FTC	0.046	0.057
Pressurizer Safety Valves	FTO	0.048	0.07
Diesel Generator, CCF of 4 DGs	FTS/FTR	0.027/0.075	Ref. is available only for 3 DGs

FTS = Fail to start
FTR = Fail to run
FTO = Fail to open
FTC = Fail to close

The distribution of sequences among these classes provides insights as to the functional failures which may dominate the risk leading to a core damage event. Table 3.1-4 of the submittal contains the designators and the description of the accident classes. The classification of the accident classes essentially follows the NUMARC Severe Accident Issue Closure Guidelines and the table also shows the NUMARC accident class designators along with those used in the Prairie Island IPE.

2.2.3.2 Human Factors Interfaces

The Human Reliability Analysis Technical Review found in Section 2.3 addresses this issue.

2.2.4 Internal Flooding

2.2.4.1 Internal Flooding Methodology

The methodology of the IPE to perform the flooding analysis consisted of three major steps:

1) Identification of potential floods and areas affected (flood zones),
2) Identification and initial screening of flooding scenarios, and
3) Quantification of important flooding scenarios.

In addition, extensive plant walkdowns supported the development of the flooding scenarios.

In the analysis, propagation of flooding to other areas (including back propagation through the drains) and isolation of the floods were considered. (Backflooding through floor drains was considered but was not credited.) Failure events which could cause flooding were the following: pipe and valve ruptures, human errors (such as errors in maintenance works), and combination of equipment failures and operations staff errors. Water spray on equipment was also taken into account, particularly when multiple systems or components were affected. Spray from a high capacity pipe was automatically included into the analysis, because breaks large enough to flood a zone were conservatively assumed to fail all equipment in that zone.

Flooding from the fire suppression systems was considered to a certain extent; i.e., if it was not bounded by a flooding from a system of higher flood capacity, such as the cooling water system.

Spurious actuation of fire suppression was not considered. The reason for this, as given in an RAI response, was the low probability of the following failure combination: failure of a fusible link in one of fire suppression spray nozzles, the nozzle must be located near essential equipment, and the spray from the nozzle must be able to fail the equipment. Potential plugging of the drains was also not considered a problem.

The flood analysis included breaks in the cooling water piping, however breaks in the component cooling water system were not considered. The reason for the neglect, also given in the RAI response, was the limited amount of the water content of the closed CC system, which is insufficient to flood the large areas where CC piping exists.

After a thorough screening process, six flood zones were retained for more detailed analysis. These were:

Screen house safeguard areas,
Screen house non-safeguards areas,
Auxiliary building 695 level,
Auxiliary building 715 level (control room chiller rooms excluded),
Auxiliary feedwater pump room in the turbine building,
Cable spreading room.

To quantify the flood initiating event frequencies the EPRI document TR-102266 was used and several assumptions regarding systems and equipment that may be disabled as a result of the flood or as a consequence of operator actions have been made.

The initiating event frequencies determined for the various flood zones are presented in Table 2, alongside with the initiating event frequencies of the internal events.

The RAI response calls the attention to the fact that the pipe break frequencies used in the flood analysis and the pipe break frequency used in the initiating event I-LOCL are different. The pipe break frequency for the initiating event represents a break anywhere in the system. The frequencies calculated for the flooding analysis represent the pipe break frequency only for piping within a specific area of the plant which has been designated a flood area.

2.2.4.2 Internal Flooding Result

The six potentially significant flooding initiating events selected by the screening process were further analyzed in the transient event tree. Examination of the reactor response established key timing for reactor conditions and operator actions and it was found that no new transient analysis was necessary to quantify flooding initiators.

Each of the flood initiated accident sequences were assigned to the following accident classes:

Flood initiated core damage early at high reactor pressure (designated as: FEH), or
Flood initiated core damage early at low reactor pressure (designated as: FLH).

The total contribution of internal flooding to the point estimate CDF was estimated to be 1.04E-05/yr, which is about 21% of the total CDF (from internal events and internal flooding). This is dominated by a single flood scenario, that accounts for almost [illegible] of the CDF due to flooding. The flood is in zone TB1 which is the Auxiliary Feedwater Pump/Instrument Air Compressor Room. This room has the main cooling water supply headers to the Auxiliary Building running through the overhead.

The flood scenario consists of a single sequence in which a large break occurs in the Loop A or B cooling water line above the auxiliary feedwater pump room in the turbine building. The resultant flood causes loss of auxiliary feedwater pumps, loss of all instrument air compressors and loss of main feedwater due to loss of instrument air and loss of lube oil cooling. Secondary cooling fails due to failure of AFW and MFW. Short term RCS inventory fails due to loss of pressurizer PORVs which fail closed on loss of instrument air.

No other flood sequence has a significant impact on the total CDF.

2.2.5 Core Damage Sequence Results

2.2.5.1 Core Damage Frequency By Initiating Events

The IPE point estimate for the core damage frequency from internal events and internal flooding is 5.0E-05/yr for Unit 1 and 5.1E-05/yr for Unit 2. Accident initiators and their percent contribution to the CDF for both units, Unit 1 and Unit 2 are listed in Table 6. As it can be seen, the core damage contributions by initiating event of Unit 2 closely resemble those of Unit 1. This is because there are only few and minor asymmetries in the designs and corresponding risk models of the units and the Unit 2 Level 1 results were obtained essentially from requantifying the Unit 1 model and replacing appropriate Unit 1 component failures by their counterpart for Unit 2.

Since new and significant safety insights were not obtained from the Unit 2 analysis, the submittal and consequently the present review discusses primarily the CDF results obtained for Unit 1.

Table 7 reproduces the CDF results in term of grouped accident types and their percent contribution. One can see, the largest contribution is due to the transient group (including the LOOP events). LOCA and internal flood events are the next contributors and they contribute almost equally. The SGTR's contribution is quite significant.

2.2.5.2 Dominant Accident Classes and Accident Sequences

The CDF results were obtained in the form of functional sequences.. Therefore the submittal used those screening criteria for reporting, which were required for such sequences in Generic letter 88-20. In fact, Prairie Island "went one step further in reporting requirement by equating accident classes with functional sequences"; i.e., it grouped core damage sequences together according to their similarity in regard to initiators, timing of core melt and effect on containment pressure at the time of the core melt and treated such a group as a combined functional sequence. In Table 3.4-1 of the submittal all the accident classes (including those that meet the reporting criteria and those that are beyond them) constituting the total CDF are presented (altogether 15 accident classes). The table contains also the description of the dominant sequence from each accident class (except accident class "THE", from which two leading sequences were considered). These dominant sequences cover only 36.4% of total CDF. The sequences include: five LOCAs, two SGTRs, six transients, two floods and one interfacing LOCA.

To illustrate the reporting process, Table 8 of this review presents the characterizations of the accident classes as functional sequences for those accident classes whose individual CDF is higher than 1E-06/r. yr. For completeness, the table also shows the accident class "V" describing Interfacing systems LOCA sequences. In the table the descriptions of the dominant sequences by accident class are reproduced, as well.

The SGTR contribution is relatively high due to dependence of pressurizer PORVs on instrument air, relative vulnerability of instrument air (due to success criteria and room cooling dependencies) and some simplifying conservative assumptions (not modeling pressurizer sprays, not modeling secondary steam dump, not modeling cross connect of instrument air to station air, and assuming steam generator PORVs sticking open with a probability of 1.0 in case of overfill).

2.2.5.3 Results of the Importance Analysis

The scope of the importance analysis performed in the IPE extends to the following areas: initiating events, systems and components, corrective and preventive maintenances (including test), and major operator actions. Two standard importance indicators were used: the Fussel-Vesely and the Birnbaum importances. In the submittal the results are summarized in two tables (Table 3.4-5 and Table 3.4-6) and six figures (Figure 3.4-1 through 3.4-6).

- The importance analyses performed on the initiating events essentially confirms the inference previously obtained from Tables 6 and 7, that the most important initiators at Prairie Island are: the LOOP(SBO) events, the Floods (T1FLD, SH1FLD), the LOCAs (MLOCA, SLOCA, LLOCA) and the LOCL (loss of cooling/service water) event.

- The importance analysis on the systems provided the ranking of the systems contributing the most to the total CDF as follows: 1) the AFW, 2) AC power, 3) room-cooling, and 4. the cooling water systems. Number one is the AFW, because in many accident sequences due to a variety of initiators (e.g.; LOOP, T1FLD, Loss of IA, Loss of DC) it is the only remaining means of secondary cooling. AC power is in second place; its loss causes heavy reliance on the TDAFW pump during an SBO when the MFW and MDAFW pumps are lost. Room cooling is important, because when Loss of IA or Loss of CL occur, the chilled water that supplies the room cooling of the Unit 1 480 V safeguards bus rooms, fails. (This condition is significantly ameliorated, as discussed earlier in the present TER based on the RAI response.)

- With respect to corrective maintenance, the maintenance unavailabilities associated with the train B AFW and the charging pumps are the most important: a.) The MDAFW pump train, because it is the most reliable pump to supply AFW to Unit 1 (the TDAFW pump has higher failure rate and the MDAFW pump of Unit 2 requires operator action to be used), and b.) The charging pumps, because after an SGTR, charging is a requirement for successful RCS cooldown and depressurization.

- With respect to preventive maintenance and test, it was found that among the unavailabilities of these types the most important is the one which is associated with the IA compressors. This is due to a combination of two negative factors: the 2/3 success criterion of the IA system (with one compressor in maintenance, a failure of the second compressor causes failure of the system) and the steady use of the compressors which requires frequent preventive maintenances whose total time rapidly accumulates to a large value.

Table 6 Core Damage Frequency by Initiating Event

Initiating Event	CDF from Initiating Event (per reactor year) Unit 1	% of Total CDF from Initiating Event Unit 1	CDF from Initiating Event (per reactor year) Unit 2	% of Total CDF from Initiating Event Unit 2
I-TR1	6.4E-07	1.3	6.6E-07	1.3
I-TR2	2.9E-08	0.06	3.1E-08	0.06
I-TR3	1.2E-06	2.4	1.2E-06	2.4
I-TR4	5.2E-07	1.0	5.5E-07	1.1
I-LOCC	5.5E-07	1.1	5.5E-07	1.1
I-LOCL	6.4E-07	1.3	6.4E-07	1.3
I-LOCA	2.2E-06	4.4	2.2E-06	4.3
I-LOCB	4.6E-07	0.9	4.8E-07	0.9
I-INSTAIR	3.2E-06	6.3	3.2E-06	6.2
I-LOOP	1.1E-05	21.2	1.1E-05	22.4
I-MSLB	*	*	*	*
I-MFLB	*	*	*	*
I-SLOCA	4.1E-06	8.2	4.2E-06	8.2
I-MLOCA	4.6E-06	9.3	4.6E-06	9.1
I-LLOCA	3.7E-06	7.5	3.8E-06	7.3
I-SGTR	6.6E-06	13.2	6.6E-06	13.0
I-T1FLD	1E-05	21	1.04E-05	20.4
I-T13FLD	*	*	*	*
I-AB7FLD	8.5E-10	2E-03	1.5E-09	2.9E-03
I-AB8FLD	*	*	*	*
I-SH1FLD	4.1E-07	0.8	4.1E-07	0.8
I-SH2FLD	4.3E-10	9E-04	5.6E-10	1.1E-03
V	2.3E-07	0.5	2.27E-07	0.5

Table 7 Accident Types and Their Contribution to the CDF

Initiating Event Group	Contribution to CDF (/r. yr)	%
LOCAs	1.22E-05	24.3
Steam Generator Tube Rupture	6.6E-06	13.2
Interfacing System LOCA	2.27E-07	0.5
Transients (w/o LOOP)	9.44E-06	18.8
Anticipated Trans. w/o Scram	3.2E-07	0.6
Internal Flooding	1.04E-05	20.7
LOOP (with SBO)	1.1E-05 (3.1E-06)	21.9 (6.2)
TOTAL CDF	5.02E-05	100.0

Table 8 Leading Accident Classes and Core Damage Sequences by Accident Class

Accident Class Designator and Description	Accident Class and Dominant Sequence		Accident Class Characterization and Dominant Sequence Description
	CDF (per r. year)	% of Total CDF	
FEH-TB1 - Flood with early core damage at high RCS pressures	1E-05	21	This accident class includes only one sequence. Flood in the AFW pump room from a break of a cooling loop header. All AFW pumps fail along with all IA compressors. MFW fails due to closure of main feed regulating and bypass valves due to loss of IA and loss of lube oil cooling to the MFW pumps. Feed and bleed fails due to loss of IA. See also Section 2.2.4.2 of the submittal.
THE - Transient with early core damage at high RCS pressures	1E-05	20	The sequences in this class are characterized by a loss of MFW through loss of IA/LOOP/SBO. Subsequently, the AFW is lost partially or completely. RCS short-term inventory fails due to loss of feed and bleed cooling caused through failure of the pressurizer PORV or the operator.
Dominant Sequence	*4.4E-07*	*8.9*	Loss of IA causes Rx trip due to loss of MFW. RCP seal cooling is successful but 11, 12 and 22 AFW pumps fail to run so 21 AFW pump cannot be used for Unit 1. Feed and bleed fails due to loss of IA and level restoration of MFW is unsuccessful.
SLL - Large/Medium LOCA with late core damage at low RCS pressures	8.3E-06	16.6	This class includes medium (break size range: 5-12 inches) or large (break size ≥ 12 inches) LOCAs in which short term inventory using SI or RHR is successful but the operators fail to switch to recirculation before the RWST is depleted causing injection to fail leading to core uncovery.
Dominant (Large LOCA) Sequence	*2.5E-06*	*5*	Successful short-term RCS inventory. Long term RCS inventory fails due to operator error in lining up for recirculation.
Dominant (Medium LOCA) Sequence	*2.2E-06*	*4.3*	Successful reactor trip and short-term RCS inventory. Long term RCS inventory fails due to operator error in lining up for recirculation.

Table 8 Leading Accident Classes and Core Damage Sequences by Accident Class

Accident Class Designator and Description	Accident Class and Dominant Sequence		Accident Class Characterization and Dominant Sequence Description
	CDF (per r. year)	% of Total CDF	
SEH - Small LOCA with early core damage at high RCS pressures	8.3E-06	16.4	The sequences within this class are characterized by either a small LOCA (break size range: 0.375 - 5 inches) or RCP seal LOCAs caused by a LOOP, loss of train A DC power or loss of cooling water. Secondary cooling using either MFW or AFW is successful but short-term RCS inventory fails due to failure of the SI system.
Dominant Sequence	*6.3E-07*	*1.3*	Loss of cooling water causes reactor trip due to loss of CC to the RCP motors. Loss of CL causes loss of chilled water and thus loss of room cooling to the 480 V safeguards bus rooms. This results in the 480 V AC bus failure causing loss of all charging pumps leading to an RCP seal LOCA that cannot be integrated by the SI pumps as they have lost CC cooling to their lube oil coolers. Local operator action to restore cooling water and 480 V bus room cooling also fail.
GLH - SGTR with late core damage at high RCS pressure	6E-06	12	The accident class is characterized by SGTR in which reactor trip, secondary cooling, short-term RCS inventory and ruptured SG isolation are successful. The operator then fails to cooldown and depressurize the RCS before the ruptured SG overfills. Relief sticks open and the operator fails to cooldown and depressurize the RCS to RHR shutdown cooling (SDC) temperature and pressure before RWST depletion occurs which causes loss of SI injection and subsequent core damage.
Dominant Sequence	*1.1E-06*	*2.1*	SGTR with operator failing to cooldown and depressurize the RCS before ruptured SG overfill. The relief sticks open followed by the operator failing to cooldown and depressurize the RCS to RHR SDC temperature and pressure before RWST depletion.

Table 8 Leading Accident Classes and Core Damage Sequences by Accident Class

Accident Class Designator and Description	Accident Class and Dominant Sequence		Accident Class Characterization and Dominant Sequence Description
	CDF (per r. year)	% of Total CDF	
BEH-NOPWR SBO with early core damage at high RCS pressures	2.8E-06	5.6	Sequences within this class are characterized by an SBO with successful TDAFW pump. The operator is successful in cooling down and depressurizing the RCS with the SG PORVs to minimize RCP seal leakage. Then he fails to restore both offsite and onsite AC power before core uncovery.
Dominant Sequence	*2.3E-07*	*0.5*	LOOP and subsequent common cause failure of four DGs. To AFW pump runs for 2 hours before batteries are depleted and SG level instrumentation is lost. The operator is successful to depressurize the SGs with SG PORVs to reduce RCP seal leakage but he fails to restore offsite and onsite AC power at 5 hours.
SLH - Small LOCA with late core damage at high RCS pressures	2.4E-06	4.8	This class can be characterized by a small LOCA (break range: .375 - 5 inches) in which secondary cooling and short-term RCS inventory using SI pumps is successful. The operator then fails to cooldown and depressurize the RCS to allow use of RHR shutdown cooling before RWST depletion occurs. High head recirculation fails due to equipment failure or operator action. The core uncovers and is damaged due to loss of makeup capability.
Dominant Sequence	*3.5E-07*	*0.7*	Small LOCA, successful Rx trip, secondary cooling and short-term RCS inventory. RCS cooldown and depressurization to RHR SDC conditions is also successful but the CC valves to the RHR heat exchangers fail to open failing RHR recirculation. Local attempts at recovery are unsuccessful.

Table 8 Leading Accident Classes and Core Damage Sequences by Accident Class

Accident Class Designator and Description	Accident Class and Dominant Sequence		Accident Class Characterization and Dominant Sequence Description
	CDF (per r. year)	% of Total CDF	
V - Interfacing Systems LOCA	2.3E-07	0.5	This class consists of interfacing systems LOCAs with a bypassed containment. ISLOCA pathways include: a) RHR to RCS loop B return line; isolation configuration: 2 check valves and a normally closed MOV, b) RHR suction from loops A and B, isolation configuration: 2 normally closed MOVs, c) Reactor vessel low-head injection line; isolation configuration: 2 check valves.
Dominant Sequence	*5.5E-08*	*0.1*	Catastrophic failure of both of RHR series loop A suction isolation motor valves followed by the failure of both of the RHR pump seals causing small LOCA outside containment. The operator is unsuccessful in cooling down and depressurizing the RCS before RWST depletion.

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 Prairie Island 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. However, while a fairly broad range of failure to restore events was modeled in the fault trees, miscalibration events were not explicitly modeled. Instead, miscalibration of various sensors and instruments was "treated through the inclusion of common cause failure modeling for the sensors or instruments themselves." In other words, the common cause failure probability for a group of related instruments was assumed to include the potential failure due to common cause miscalibration. The submittal argues that because of redundancy in instrumentation for activation of safety systems, common cause failures (and miscalibrations) of sensors were determined to be insignificant contributors to plant

risk." While there are reasonable aspects of such an argument, it should be noted that miscalibration events have been explicitly modeled in other IPEs and in some instances have been shown to be significant contributors. Thus, the licensee's treatment of miscalibration events may have precluded identification of important pre-initiator events and is therefore a weakness of the HRA.

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

In the licensee's response to an NRC request for additional information (RAI), it was indicated that errors committed during corrective and preventative maintenance and testing (including those during refueling outages) were considered and their applicability was evaluated during development of the fault trees by reviewing operating procedures, maintenance procedures, administrative control documents, and EOPs. Testing and maintenance procedures were reviewed to determine if any component is effectively disabled and cannot automatically be restored or realigned on an appropriate initiating signal. If so, then a basic event was added to the fault tree to reflect the potential for failure to restore. However, if an indication of the misalignment is given in the control room, the event was not modeled. If some other indication is given in areas toured by operators during rounds, the error was only considered for the interval between rounds.

While no explicit statements regarding discussions with plant personnel on the interpretation and implementation of procedures were provided, it appears that relevant information sources were examined and that factors which could influence the probability of pre-initiator restoration failures were considered.

2.3.1.3 Screening Process for Pre-Initiator Human Actions

The licensee stated that no screening values were used when modeling pre-initiator human errors. As will be discussed in the next section, values were calculated for all the restoration events that were modeled. For pre-initiator events included in the fault trees that were later determined to result in indications occurring in the control room, their failure probabilities were set to zero and they were left in the models "for possible use in sensitivity studies."

2.3.1.4 Quantification of Pre-Initiator Human Actions

Restoration of components with maintenance procedures calling for post-maintenance verification of component status were assigned HEPs of 0.003. A failure to restore probability of 0.01 was assigned to components without such a verification. Plant-specific component unavailabilities were then calculated using the HEPs in conjunction with factors such as maintenance frequency and duration, test frequency and interval, refueling outage frequency, and time from completion of the corrective maintenance to the retest of the component. The unavailability equation was indicated as being from ASME paper 91-JPGC-NE-11 and appeared reasonable. The basis for the selected HEPs was not provided, but the values are reasonable and consistent with those that would be obtained using the method described in NUREG/CR-4772 (ASEP) for quantifying pre-initiators. Given plant practices regarding maintenance of separate trains, all restoration failures were considered independent events. Dependencies associated with miscalibrations were at least indirectly covered by grouping instruments according to potential common cause failure mechanisms. Similar instruments would probably have similar calibration procedures, but other factors such as the plant schedule for calibrations could also have been relevant.

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 Prairie Island IPE addressed both response and recovery type post-initiator human actions. Response type actions were those modeled only when clear procedural guidance (normal, abnormal, or emergency procedures) existed for the operator. Recovery actions included cases where procedural guidance may not have been available, but "the operators training or knowledge are assumed to lead him to perform the required action.." Repair activities (e.g., restoration of instrument air or component cooling water), local operation of systems (e.g., manual opening of valves), and recovery of offsite power or a DG are examples of recovery events modeled in the IPE. Eleven recovery events are described in the submittal and none appeared to require extraordinary behavior on the part of the operators.

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

A detailed discussion of the initial process for the identification and selection of post-initiator human actions was not provided by the licensee. However, it is noted in the submittal that "only those post-accident operator actions required to initiate systems... were included in the fault trees." These actions were described as being based on normal, abnormal, and emergency procedures. Operator actions in response to equipment failures were not modeled until after the sequence cutsets were generated. If more than one operator action occurred in the same cutset, "either independence of the human actions was confirmed, or a change was made to correctly model dependence between human errors." The submittal also indicated that in the "human factors review performed in support of the HRA," walkdowns of selected local and control room panel operator actions were performed to verify assumptions and to "look for factors not previously considered." In addition, control room personnel were interviewed to discuss roles and responsibilities during actions, timing of actions, and performance of specific actions. Thus, it appears that activities were conducted that would to help ensure appropriate modeling of operator actions.

2.3.2.3 Screening Process for Post-Initiator Response Actions

The submittal indicates that screening values were assigned "by following a flow chart and answering a series of questions." Neither the questions nor the flow chart was provided, but "plant-specific estimates

of the time available to initiate and perform the action" and "PSFs associated with degree of difficulty and stress" were considered. The screening HEPs were based on the Handbook (NUREG/CR-1278), Wash-1400 (NUREG-75/014), and "the data sources used in the IDCOR BWR IPE methodology." Thus, at a minimum a systematic approach was taken in the assignment of post-initiator screening values. Examples of typical operator actions and their associated screening values were presented in Table 3.3-10 of the submittal. While some of these values (e.g., 1.0E-4 and 1.0E-3) might normally be considered non-conservative for screening values, they may be justified given the apparent detailed screening analysis and the licensee's consideration of dependencies (as is discussed further below). As noted above, the licensee indicated that if more than one operator action occurred in the same cutset, "either independence of the human actions was confirmed, or a change was made to correctly model dependence between human errors." Furthermore, in response to an additional NRC RAI on the consideration of dependencies, the licensee stated that each operator action was examined when it was calculated, to determine which operator actions were dependent due to the same cognitive process. Actions determined to use the same cognitive process were given large screening values to ensure that they were not truncated from the final core damage sequence equation. The final values of these operator actions were later set to values that appear in the relevant tables in the submittal. The licensee also noted that all but six of the seventeen human actions left at screening values did appear in the final CDF or containment failure sequence equations. The remaining six actions were said to be unlikely to ever appear in risk significant sequences, since they must fail in conjunction with highly reliable components. Thus, the licensees screening approach appears to have precluded inappropriate truncation. The licensee also noted that the HEPs obtained for the human actions receiving detailed analysis were "at about the same value or lower than the screening analysis results."

2.3.2.4 Quantification of Post-Initiator Human Actions

Post-initiator response type actions found to be important received detailed HEP development. An action was important if it contributed significantly to baseline core probability or "if a change in the failure rate could cause significant increase in overall core damage probability." Most of the actions receiving detailed analysis were quantified with the ASEP method (NUREG/CR-4772). The response to the NRC RAI states that a more refined estimate for the five most important human actions was obtained using the THERP Handbook (NURG/CR-1278). The five actions quantified with the Handbook included actions associated with transferring to recirculation following a LOCA and cooling down and depressurizing during a steam generator tube rupture. The values for the transfer to recirculation ranged from 0.0012 to 0.008 and the values for cooldown were 0.011 and 0.0065. While none of these values are grossly inconsistent with values for similar events in other IPEs, it did appear that the switchover to recirculation HEPs could be optimistic given the available time frames for the small, medium, and large LOCA scenarios. A description of how these values were obtained using the Handbook methodology was not provided in the submittal or in the licensee's response to the initial RAI. The submittal does note that the analysis was similar to that used in ASEP, except that more detailed analysis is performed for types of procedures used (with or without sign-offs), verification steps, crew size and crew response timing, errors of omission and commission, timing, expected stress and dependencies between operator actions.

In response to an additional NRC RAI regarding the events quantified with the THERP Handbook, the licensee provided detailed calculation sheets illustrating how the HEPs were obtained for the switchover to recirculation events. While the analyses were reasonable, the licensee apparently took some credit for diagnosis of a LOCA and the eventual need for switchover prior to the occurrence of the low RWST level alarm. This alarm is apparently the primary cue for the switchover action. The THERP annunciator response model was used to determine the post RWST level alarm HEP and the time based cue response

model was used to determine the HEP for diagnosis prior to the occurrence of the critical cue. The analysts assumptions were not unreasonable and the approach was consistent with the application of THERP. Given other conservatisms noted by the licensee in the analysis of these events (e.g., conservative estimates of time available), the obtained HEPs appear reasonable.

A description of the application of the ASEP methodology to the other events appeared sound. A review of the obtained values suggested a tendency toward conservatism, as was noted by the licensee and as is generally the case with applications of ASEP (see Table 3.3-3 of the submittal). The licensee also noted that a relatively conservative diagnosis time was assumed for feed and bleed and that the same time was applied in all instances of the events quantification. This approach has the potential to lead to pessimistic HEPs for some cases of feed and bleed.

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, such as the ASEP methodology. In the licensee's discussion of the application of the method, it is clear that appropriate timing parameters were considered. MAAP runs were used to determine the latest time an operator action could be completed and guidance from ASEP was used to assess the time needed to complete actions in control room. For action times outside the control room, an engineer and former SRO was consulted and confirmatory walkdowns were performed in some cases. The time at which a compelling signal is received was considered and diagnosis times were computed. Examples indicated appropriate use of the timing parameters within the application of the method.

2.3.2.4.2 Other Performance Shaping Factors Considered

As noted above, the application of the Handbook to five important human actions involved detailed consideration of a set of important PSFs. The PSFs addressed in the application of ASEP included training, practice during simulator training, and whether the event was covered in the EOPs. In addition, the existence of written procedures for conducting the action, whether the procedural actions were step-by-step or dynamic, stress level, size of crew and time available for recovery credit were considered. These PSFs are those normally considered in applying the ASEP methodology.

The submittal also stated that the "control room design review" was reviewed and that walkdowns of selected operator actions and control room panels were made to look for factors not previously considered. Whether and how these factors influenced resulting HEPs outside of the basic treatment of PSFs by the HRA methods was not discussed.

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 was treated in using the ASEP quantification approach. The licensee's response to the RAI also noted that the same timing parameters were used in applying the handbook method.

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. Although not explicitly discussed, the licensees clear evaluation of timing factors and consideration of dependencies appeared to cover this type of dependence.

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. Several discussions in the submittal and in the licensee's responses to the RAIs indicate that potential dependencies among the operator actions were appropriately considered.

2.3.2.4.4 Quantification of Recovery Type Actions

The licensee's response to the RAI indicated that all recovery probabilities (with a couple of exceptions) were derived from NSAC-161, "Faulted Systems Recovery Experience" and that recovery actions were only added to cutsets when it was apparent that an operator would have sufficient time to perform the additional action. Local recovery of valves which failed to open or close was credited only when there was control room indication of valve position and the valve was easily accessible . A non-recovery probability of 0.025 was assigned to these events. Recovery of equipment such as pumps had the same criteria, but the non-recovery probability was "approximately 0.5."

The validity of the derivation of HEPs for recovery actions from NSAC - 161 was addressed in an additional RAI to the licensee. The RAI suggested that the nature of the grouping of different recoveries in NSAC-161 is fairly "coarse." And that in many cases the grouping appears to be "mixing apples and oranges." A discussion of the derivation process for recovery HEPs (with examples) was requested. In response, the licensee demonstrated that a thoughtful analysis of the data in NSAC - 161 was conducted and that an attempt was made to use the most appropriate information. While it could still be argued that the data in NSAC - 161 is insufficient in many cases for the derivation of recovery HEPs for PRA use, the examples provided by the licensee suggested that the data was used conservatively.

2.3.2.4.5 Human Actions in the Flooding Analysis

Operator actions were apparently not quantified in the Prairie Island IPE. However, the submittal states that "the minimum time allowed for a zone to flood was based on the indications that the operators would receive to alert them to the flood." Operators were assumed to find and isolate the source of flooding in 20 minutes if an alarm or other indication would tell them exactly where the flooding was occurring. If the indication occurs, but does not tell them exactly where the flood is located, then 60 minutes is assumed before they will isolate the leak. If no indication is given, it was assumed that the operators would find and isolate the flood during their rounds within three hours.

No other discussion of operator actions in regard to the flooding analysis was found.

2.3.2.4.6 *Human Actions in the Level 2 Analysis*

A review of the Prairie Island Level 2 analysis failed to find any discussion of the quantification of operator actions. Apparently RCS depressurization was discussed, but the related operator action and its quantification was not addressed.

2.3.2.5 Important Human Actions

The Prairie Island IPE provided a list of important human actions as determined on the basis of a Fussel-Vesely analysis. Events identified as accounting for more than 1% of the total CDF and their HEPs are presented below in Table 9. Each event's percent contribution to CDF is also listed.

A review of the HEPs associated with each event did not identify any events with an HEP that would be considered inconsistent with failure probabilities obtained for similar events in different plants or that appeared to be excessively low.

Table 9 Important Human Actions

Event Description (percent contribution to CDF)	Human Error Probability (HEP)
Bleed and feed (no "S" signal) - for single operator action cutsets (7.2%)	0.039
Cooldown and depressurize RCS to stop tube leak *before* SG overfill (6.7%).	0.011
Transfer to recirculation during large LOCA (5.0%)	0.0084
Transfer to recirculation during medium LOCA (4.3%)	0.0027
Open doors on loss of room cooling (3.4%)	0.067
Cross tie to unit 2 motor driven AFW pump (3.2%)	0.032
Cooldown and depressurize RCS to stop tube leak *after* SG overfill (2.2%).	0.0065
Transfer unit 2 AC to unit 1 during unit 1 SBO (1.3%).	0.0032
Bleed and feed (no "S" signal) - for multiple operator action cutsets (1.7%)	0.071

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 14 accident classes (ACs). Definition of the accident classed is discussed in Section 4.3 of the IPE submittal. The parameters used in the IPE to define the ACs include:

1. Accident initiator,
2. Core melt timing,
3. RCS pressure at the time of core melt.

The accident initiators for the ACs analyzed in the Prairie Island IPE include transient, station blackout (SBO), LOCA, SGTR, ISLOCA, and ATWS. The timing of core melt depends primarily on whether core melt is caused by injection (early) or recirculation failure (late), and RCS pressure, which is defined in the IPE as the primary system pressure being high enough to entrain the core debris out of the cavity upon vessel failure, depends on the type of accident initiators. For example, except for the accident classes involving medium and large LOCA sequences, the RCS pressure for the accident classes involving other accident sequences is high.

The conditional probabilities of the ACs for the various accident initiators are: 23% for small LOCA (2 ACs), 22% for transient (2 ACs), 19% for internal flooding (2 ACs), 16% for medium or large LOCA (2 ACs), 13% for SGTR (2 ACs), 6% for SBO (1 AC), 0.6% for ATWS (2 ACs), and 0.4% for ISLOCA (1 AC). The most probable AC is THE (20% CDF), an AC with accident sequences initiated by transient initiators with early core melt and high RCS pressure. This is followed by FEH (19% CDF), an AC initiated by internal flooding, with early core melt and high RCS pressure.

Unlike the PDSs used in some other IPEs, the availability of containment systems are not explicitly included in the definition of the ACs in the Prairie Island IPE. In the Prairie Island IPE, containment system fault trees (for containment spray injection, containment spray recirculation, and containment fan coil units) are quantified as frontline systems, along with the Level 1 frontline and support system fault trees, using linked fault tree models. The containment systems fault tree cutsets are input to the CET branches as necessary to support CET quantification.

The ACs defined in the Prairie Island IPE to provide front-end back-end dependencies for the Level 2 analysis of the IPE seem adequate. Although their definition does not include the parameters indicating the availability of containment systems, the use of fault tree linking to determine their availability in CET quantification seems adequate.

2.4.1.2 Containment Event Tree Development

Probability quantification of severe accident progression is performed using containment event trees (CETs). The development of the CETs is discussed in Sections 4.5 of the IPE submittal. Five CETs are developed in the Prairie IPE for the 14 accident classes obtained in the IPE. The CETs includes the following top events:

1. Accident class,
2. Bypass due to SGTR,
3. Containment isolation,
4. In-vessel recovery,
5. Reactor depressurization,
6. Early containment challenges,
7. Ex-vessel injection,
8. Containment pressure control,
9. Fission product scrubbing - Containment Spray.

Figures 4.5-1 through 4.5-5 of the submittal show the structures of the five CETs. Figure 4.5-2, the CET for SBO accident class with early core melt, includes an additional top event to address the probability of ac power

recovery. The CETs in the Prairie Island IPE have from 22 (for medium or large LOCA with late core melt) to 49 end states (for SBO with early core melt). Since containment failure is assured for ISLOCA and SGTR accident sequences (i.e., by containment bypass), CETs are not developed for the ACs associated with these sequences and all of the them are assumed in the IPE to lead to containment bypass. In general, the CETs developed in the Prairie Island 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.

The quantification of the CET in the Prairie Island IPE is based on plant-specific phenomenological evaluations. The evaluations include modeling and bounding calculations (based upon experimental data), consideration of phenomenological uncertainties, and MAAP calculations. 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.

Containment Failure Modes not Included in CET Quantification

Containment failure modes are discussed in Section 4.4 of the IPE submittal. Although all important severe accident containment failure modes that are discussed in NUREG-1335 are addressed in the IPE submittal, some of them are ignored and not evaluated in CET quantification. These include those associated with the melt-through of the containment steel shell (i.e., liner melt-through), vessel thrust forces, and thermal attack of containment penetrations.

The potential failure of the steel shell due to direct contact with molten corium is discussed in the IPE submittal and also in the licensee's response to the RAI (Level 2 Question 5). The potential of containment shell melt-through exists only during high pressure melt ejection (HPME) when a significant amount of core debris is dispersed to the outside of the reactor cavity and comes into contact with the steel shell structure of the containment. Two debris dispersion paths are identified and evaluated in the IPE. The first one is through the instrument tunnel to the upper compartment, exiting the cavity at the seal table structure; the second one is through the access hatches from the containment to the instrument tunnel. Of the two paths, the second one presents a more significant challenge to containment integrity. The access hatches to the instrument tunnel are in an open area on the basement level of the containment, and for both of the Prairie Island units, one of the two hatches faces toward the steel containment, about 30 feet away, with a largely unobstructed path in between. According to the licensee's response to the RAI (Level 2 Question 5), a scoping study performed in the IPE shows that the temperature generated by the debris adhering to the steel wall is insufficient to melt the steel and breach the containment even with conservative assumptions about the mass, distribution, and heat generation of the debris expelled from the instrument tunnel. However, details of the scoping study are not provided in the submittal and the potential effect of corium attack on reducing containment pressure capability is not discussed.

The probability of containment failure due to vessel thrust forces and thermal attack of containment penetrations are discussed in Section 4.4.3 of the IPE submittal. For thrust forces, an estimate of the thrust force generated during HPME is obtained and compared with the weight of the reactor vessel and the capability of the primary shield wall acting as a restraint to determine the probability of containment failure due to this phenomenon. For thermal attack, the penetrations and the seal materials used for the penetrations are identified and evaluated against representative severe accident temperature profiles to determine the potential of containment failure due to excessive leakage from thermally degraded seals. Results obtained in the IPE show that the probabilities of containment failure from both of the above phenomena are negligible.

Containment Failure Modes Considered as Unlikely but Included in CET Quantification

The containment failure modes that are considered as unlikely in the IPE but are assigned a small failure probability in CET quantification include those associated with direct containment heating (DCH, 1E-3), in-vessel steam explosion (Alpha mode failure, 1E-4), ex-vessel steam explosion (1E-3), and hydrogen combustion (5E-3). Containment isolation failure is also considered as unlikely but is evaluated in the CET quantification using the data obtained in the containment isolation analysis (see Section 2.4.1.4 of this TER).

In the IPE, deterministic evaluation of the expected peak pressures was performed for the phenomena that would cause containment pressure challenges (e.g., DCH and hydrogen burn). The estimated pressure loads were then compared with the containment fragility curve to determine the probability of containment failure. For those phenomena with other containment challenges (e.g., steam explosion), a combination of deterministic analyses and expert opinion found in the open literature is used to estimate containment failure probability. Because of the uncertainties associated with these phenomena, the values used in the IPE for these phenomena are described as qualitatively evaluated values (in the licensee's response to RAI Level 2 Question 3). According to the discussion provided in the IPE submittal the values used in the IPE for these phenomena seem to be adequate.

Containment Bypass and Induced Steam Generator Tube Rupture (ISGTR)

Containment bypass is considered in the IPE as one of the dominant containment failure modes (Section 4.4.2). According to the Prairie Island IPE, containment bypass occurs for the SGTR and ISLOCA events, it also occurs with an induced creep rupture of the steam generator tubes. As mentioned above, CETs are not developed for SGTR and ISLOCA accident classes and containment bypass is assured for these accident classes. Their contributions to the total CDF are 13% and 0.4%, respectively. On the other hand, ISGTR is evaluated in the CET as one of the CET top events (Section 4.5.3.1). According to the IPE, ISGTR occurs only for cases with the RCS at high pressure, with dry steam generators, and with the restart of the reactor coolant pumps (RCPs). Although the probability value used in CET quantification for ISGTR is not provided in the submittal, results presented in the submittal show a significant contribution from ISGTR (about 31% of CDF).

The high ISGTR probability obtained in the IPE is primarily due to the procedures that call for the restart of the RCPs under severe accident conditions. According to the licensee's response to the RAI (Level 2 Question 7), these emergency procedures have been changed to prohibit the restart of a reactor coolant pump with a dry steam generator under severe accident conditions (based on recommendations from the Westinghouse Owners Group). This procedure change is expected to reduce significantly the probability of ISGTR. Because of the change of the EOPs, the C-Matrix provided in the licensee's response to the RAI (Level 2 Question 7) does not include the contribution from ISGTR.

In-Vessel Recovery

Two means for in-vessel recovery are considered in the IPE: restoration of injection and ex-vessel cooling of in-vessel core debris. According to the IPE, the recovery of core injection has minor effect on preventing vessel failure (i.e., in-vessel recovery). This is because of the short time available for injection recovery for the core melt sequences in which core damage is caused by the loss of injection (i.e., the early melt sequences). On the other hand, external cooling is assumed very effective in preventing vessel failure for core melt sequences in which the RWST water has been injected into the containment (and thus the lower vessel head is submerged). In the IPE, the probability of lower head failure is assumed to be 0.1 if the RWST has been injected. Since significant uncertainty is associated with this probability value, it is included in the sensitivity

study. Results of the sensitivity study shows that containment failure probabilities are not significantly affected by the probability value used for this vessel cooling mode.

Although the sensitivity studies performed in the Prairie Island IPE shows little effect of ex-vessel cooling on containment failure, source terms obtained from containment failure may be affected by ex-vessel cooling because fission product production and release path are modified (e.g., in-vessel release from a dry debris versus ex-vessel release from a debris bed covered by water). Since ex-vessel cooling is not included in the MAAP model which is used in the IPE for source term calculation, the source terms obtained for the cases with low pressure vessel breach are used for cases with no vessel failure. It is argued in the licensee's response to the RAI (Level 2 Question 1) that this is a conservative approach because the former involves the release of non-volatile fission products from core concrete interaction. However, no quantitative information is provided in the response to support this argument.

According to the data provided in the C-Matrix (provided in the licensee's response to the RAI, Level 2 Question 7), the probability of in-vessel recovery (i.e., no vessel failure) is over 99% for late core melt sequences and zero for early core melt sequences. The frequency of all ACs that involve in-vessel recovery is about 23% of total CDF. The primary contributor to this probability is medium or large LOCA with late core melt (15% CDF). This is followed by small LOCA (6% CDF) and transient (2% CDF).

RCS Depressurization

RCS depressurization is one of the CET top events. According to the submittal, RCS depressurization from the operation of both the pressurizer PORVs and the steam generator PORVs are credited in the IPE. However, because of time constraint, RCS depressurization by the above mechanisms is assumed to be successful only for late melt cases when vessel lower head penetration is delayed by ex-vessel cooling. The conditional probability of all late failure sequences in which RCS depressurization may be effective is about 8% (of CDF). The primary contributor is small LOCA (6% CDF). This is followed by transient (2%) and ATWS (0.4%).

In addition to the above two mechanisms, RCS depressurization by creep rupture of the RCS system is also credited in the IPE. In the CET quantification, this RCS depressurization mechanism is considered to be effective only if the core can be retained in the vessel for an extended period of time such that the RCS pressure boundary can heated to temperatures that can cause creep rupture. It is assumed in the IPE that this is effective only if the lower vessel head is submerged, and RCS depressurization is almost assured under this condition (a probability value of 0.99). Because of uncertainties associated with this mechanism, it is included in the sensitivity studies of the IPE.

As discussed above, in the Prairie Island IPE model, both in-vessel recovery (i.e., no vessel failure) and RCS depressurization are effective only for cases with the vessel lower head submerged, occurring in accident classes with late core melt. For these accident classes, the probability of in-vessel recovery (i.e., no vessel failure) is over 99%. Since there is no vessel lower head failure for these cases, RCS pressure is not significant and thus not reported in the IPE submittal. For the remaining 1% of these accident classes, the RCS pressure is low. Therefore, for late core melt sequences, the RCS either remains intact or fails at low pressure.

Early Containment Failure

Although containment failure timing is defined in the Prairie Island IPE as relative to the declaration of a General Emergency (Table 4.3-3), early containment failure modes considered in the CET involve those

occurring before or at the time of vessel bottom head failure (p 4.5-10). The failure mechanisms considered in the CET for early containment failure include hydrogen burn before or at RPV failure, direct containment heating (DCH), in-vessel and ex-vessel steam explosions, and vessel blowdown forces. As discussed above, all of these phenomena are considered in the IPE as unlikely to cause containment failure. However, a small containment failure probability is assigned in the IPE for CET quantification.

The frequency of early containment failure is about 0.8% of total CDF. The accident class that has the highest conditional probability of early containment failure is that associated with internal flooding (1%). This is followed by medium/large LOCA (0.9%), small LOCA (0.7%), and transient and SBO (both of 0.6%).

Debris Coolability and Late Containment Failure

Late containment failure occurs if the debris discharged from the reactor vessel is not coolable or if all decay heat removal systems fail. Because of the thin debris layer expected on the containment floor following vessel failure (less than 25 cm for Prairie Island), the debris is assumed to be coolable if the RWST water is injected to the containment. Ex-vessel debris is thus assumed to be quenched if one train of any of the following three systems works : high head safety injection, low head safety injection, or containment spray.

Even with a coolable debris, decay heat removal is still required for containment pressure control. In the IPE, containment pressure control is assumed to be successful if one fan cooler unit or one train of RHR in recirculation is available. The only exception to the above requirement is that associated with high pressure melt ejection. Since a significant amount of core debris is expected to be relocated to the upper compartment of the containment, which cannot be cooled by the water in the reactor cavity, successful operation of containment spray is required to prevent long-term containment overpressure failure for this case. According to the CET results, containment overtemperature/overpressure failure due to lack of debris cooling for the debris relocated to the upper compartment is the most significant late containment failure mode (about 17% of CDF). This is followed by that caused by the loss of all decay heat removal systems (6% of CDF), and basemat melt-through when most of the debris remains in the reactor cavity (0.1%) with no cooling water.

The conditional probability of late containment failure is about 23% of total CDF. The accident class that has the highest conditional probability of late containment failure is small LOCA (58%). This is followed by transient (39%) and SBO (7%). The conditional probabilities of late containment failure for the remaining accident classes (i.e., medium/large LOCA, IFL, and ATWS) are less than 1%. Since the availability or recovery of the containment systems are obtained in the CET quantification by fault tree linking and detailed information is not provided in the IPE submittal or the licensee's response to the RAI (Level 2 question 3), the reasons for the differences in late failure probabilities for the different accident classes are not known. Since the quantification process for system availability is identical to that used in Level 1, it is expected to be consistently maintained throughout the CET analysis.

Source Term Scrubbing

Credit is taken in the IPE for fission product scrubbing by the operation of containment spray. Limited credit is taken for the operation of fan cooler unit.

2.4.1.3 Containment Failure Modes and Timing

The Prairie Island containment ultimate strength evaluation is described in Section 4.4.1 of the IPE submittal. Containment failure pressures were obtained in the Prairie Island IPE by a simplified plant-specific structural

analysis using actual material failure stresses. Containment failure pressures were calculated for four locations. Uncertainties of containment failure pressures due to variability in material properties and analytical modeling were estimated to establish containment failure distributions. The assumed coefficients of variations (i.e., the standard deviation divided by the mean) for the four estimated failure pressures varied from 11% to 15%. Composite failure pressure distributions for Prairie Island were then determined from the results obtained for the four failure pressures. The median containment failure pressure obtained in the Prairie Island IPE is 150 psig.

The containment failure pressures and their distributions obtained in the Prairie Island IPE seem to be consistent with those obtained in other IPEs. For Prairie Island, a large catastrophic failure of the steel shell is assumed in the IPE if containment is failed by overpressure. This seems adequate for the free standing steel shell containment structures used in Prairie Island.

2.4.1.4 Containment Isolation Failure

Containment isolation failure is one of the top event in the Prairie Island CETs. The evaluation of containment isolation is discussed in Section 4.4.3 of the IPE submittal. Additional discussion is provided in the licensee's response to RAI (Level 2 Question 4). In the IPE, only pipes with diameters greater than 2 inches are evaluated, and results show that the probability of containment isolation failure for Prairie Island is about 5E-4 (Table 4.4-3). Because of its small probability, containment isolation failure is identified in the IPE as unlikely containment failure modes. Nonetheless, containment isolation failure is included in the CET quantification and three of the seventeen CET end states that are selected in the IPE for source term calculations involve containment isolation failure. Based on the C-Matrix provided in the licensee's response to the RAI (Level 2 Question 7), the conditional probabilities of containment isolation failure for the various accident classes vary from 0 (e.g., for transient with late core melt) to 6E-4 (i.e., for internal flooding with early core melt). The conditional probability of containment isolation failure for all accident classes is about 2E-4.

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

Both primary and secondary depressurizations are considered in the CET. These involve the opening of the primary and secondary PORVs by operator actions. The probability of successful RCS depressurization is evaluated as part of the fault tree linking and the probability values used in the CET quantification are not provided in the IPE submittal. Since RCS depressurization is considered only for accident classes with late core melt and high RCS pressure, which contributes only about 8% of total CDF, the effect of RCS depressurization on CET quantification is not expected to be significant.

Recovery of ac power is included in the SBO CET as one of the top events. Additionally, availability and/or recovery of containment systems may also be considered in the IPE analyses[1]. The probability values used in

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[1] Although repairs of failed equipment is mentioned in the IPE submittal (e.g., p 4.5-12), recovery of containment systems probably is not credited in CET quantification. It is stated in the response to one of the RAI questions (Level 2 Question 11) that "No credit for repair and recovery of containment spray recirculation was given in the baseline IPE analysis". Since the recovery of this system would have the most significant effect on

the IPE for the above items are not provided in the IPE submittal because they are not estimated as point values but are obtained from equations developed for the CET event tree headings by linked fault tree models (response to Level 2 Question 3). Because they are not normal output to CET quantification and a significant amount of effort would be required to recalculate and report these values, they are not provided in the response. In general, the treatment of system availability and recovery in the CET analysis is similar and consistent to that used in the Level 1 analysis.

Another issue that is related to system recovery is the change of procedures at Prairie Island that may affect the availability of containment spray in recirculation. In the Prairie Island IPE, containment spray in recirculation mode is required to provide long-term cooling of the debris relocated to the upper compartment to prevent late containment failure. Results of sensitivity studies show that the probability of late containment failure increases significantly (from 21% to 63%) if the relocated debris is not coolable. A similar change in containment failure probability is expected if containment spray is not available. At the time of IPE the transfer to containment spray recirculation was proceduralized in the plant EOPs. However, changes of the EOPs made after the IPE has this guidance removed. The potential impact of this procedure change on CET quantification is asked in one of the RAI questions (Level 2 Question 11). The licensee's response to this question is to emphasize the long time required for containment failure by this mechanism (on the order of 3 to 4 days) such that ample time would be available to align the system for operation despite the unavailability of procedures. It also emphasizes the low source terms associated with containment failure associated with the loss of containment spray in recirculation mode.

2.4.1.6 Radionuclide Release Characterization

The end states of the CET (defined as damage states in the Prairie Island IPE submittal) are discussed in Section 4.3.2 of the IPE submittal. Three parameters are used to define a damage state. They are:

i. Reactor status,
ii. Containment status, and
iii. Containment failure timing.

Table 4.3-3 of the IPE submittal shows the values acceptable for these parameters. A total of 17 CET end states are defined in the Prairie Island IPE for source term definition. Source terms of are determined by the analyses of representative sequences using MAAP computer codes.

The CET end states defined in the Prairie Island IPE using the above three parameters are not as detailed as those defined in some other IPEs in which more parameter are considered. Although the approach used in the Prairie Island IPE makes the CET results more traceable, it may occasionally lead to the grouping of sequences which do not behave in an entirely similar way. This is realized in the Prairie Island IPE and attention is paid on the selection of the accident sequences. From the description provided in the IPE submittal it seems that the CET end state grouping for source term definition in the Prairie Island IPE is adequate.

The CET quantification results provided in Table 4.6-1 of the IPE submittal show 17 CET end states that have non-zero frequencies. Among these CET end states are three bypass end states that are obtained directly from

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preventing containment failure, the lack of consideration of recovery of this system seems to indicate that the recovery of other containment systems are not credited in CET quantification.

the accident classes (i.e., two SGTR and one ISLOCA) and two induced SGTR end states. The 12 non-bypass CET states include 4 with late containment failure, 3 with early containment failure, 2 with containment isolation failure, and three with no containment failure. Including induced SGTR[2], the percentage contributions of these CET end states to the total CDF are 32% for no failure, 23% for late failure, 1% for early failure, 45% for bypass failure. The probability of ISGTR is about 30%. The probability of bypass failure is decreased by 30% and the probability of no failure is increased by 30% if ISGTR is not included.

Source terms for the CET end states are determined by accident progression analyses using MAAP code (MAAP 3.0B Revision 19.0). Source terms obtained from the computer code calculations are presented in Table 4.7.-2 of the IPE submittal. Source terms are presented in these tables in terms of release fractions of some of the representative radionuclides (e.g., CsI).

The use of the computer code calculations for source term definition is discussed in the following section of this report.

2.4.2 Accident Progression and Containment Performance Analysis

2.4.2.1 Severe Accident Progression

Sequence selection for fission product release characterization is discussed in Section 4.7 of the Prairie Island IPE. According to the Prairie Island IPE, the purpose of sequence selection is to choose specific accident progression sequences that best approximate the representative source term results for each relevant CET end states based on the consideration of the dominant sequence in each end state and other factors that influence the source term results. The magnitudes of fission product release are taken directly from MAAP calculations. Typically, the MAAP calculations are continued for 48 hours from sequence initiation or for 24 hours following containment failure, whichever is longer. Results of MAAP calculations are shown in Table 4.7-2. According to the results presented in the table, MAAP calculations are performed for the selected sequences for 11 of the 17 CET end states. The release fractions for the remaining 6 CET end states are assigned the values obtained for the CET end states with more severe results. The sequence selection and the assignment of release fractions for source term determination seem adequate.

Because of the uncertainties in the MAAP modeling of fission product behavior and variations in the specific sequence definition, the representative source term results are further grouped to provide more general conclusions about the results. The grouping is based on the magnitudes of the release fractions of noble gases, volatile releases (characterized by CsI and CSOH releases), and non-volatile releases (characterized by the largest of the tellurium, strontium, or barium release). Six summary source term types are obtained by the grouping. Results of source term grouping shows that 31% of all core damage sequences has releases limited to those corresponding to normal leakage (Type I), and 52% has high noble gas releases, but with low (less than 1%) or low-low (less than 0.1%) volatile and non-volatile fission products releases (TYPE II).

One CET end state in Type II that is of particular interest is that associated with induced SGTR (about 30% of total CDF). The low release for this bypass sequence is due to the assumption used in the IPE regarding the steam generator valve conditions. It is assumed in the IPE that the steam generator valves, which open to

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[2] According to the licensee's response to the RAI (Level 2 Question 7) ISGTR is effectively precluded by a procedure change. The C-Matrix provided in the response does not include ISGTR.

relieve the steam generator pressure, would reclose successfully. This limits the release to a relatively short duration puff, followed by a series of shorter puffs. All releases are terminated upon vessel failure when the primary system depressurizes to containment pressure. The effect of the harsh environmental conditions on the proper operation of valves are discussed in the licensee's response to the RAI (Level 2 Question 6). However, the concern over ISGTR may be moot because, at this point, the EOP to restarts the RCPs, the primary reason for the high ISGTR probability, is changed. If the probability of ISGTR is removed, the probability of Type I release would increase from 31% to 60%, and the probability of Type II release would decrease from 52% to 22%.

The release types that have high noble gas and volatile fission products releases are Type V (14.6%) and Type VI (0.5%), both of which involve containment bypass. Type V includes SGTR as an initiating event and ISGTR with the steam generator valves not reseating properly. Type VI includes ISLOCA sequences. The sequences that involve early containment failure are grouped to Types III (0.3%) and IV (0.6%), both of which have release fractions for volatile fission products releases between 1% to 10%.

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

CET end states (or containment failure modes) and their frequencies obtained from the Prairie Island CET quantification are discussed in Section 4.6 of the submittal. Table 10, below, shows a comparison of the conditional probabilities for the various containment failure modes obtained from the Prairie Island IPE with those obtained from the Surry and Zion NUREG-1150 analyses.

Table 10 Containment Failure as a Percentage of Total CDF

Containment Failure Mode	Prairie Island Nuclear Generating Station IPE++	Surry NUREG-1150	Zion NUREG-1150
Early Failure	0.8	0.7	1.4
Late Failure	22.6	5.9	24.0
Bypass	44.7	12.2	0.7
Isolation Failure	0.02	*	**
Intact	31.8	81.2	73.0
CDF (1/ry)	4.9E-5	4.0E-5	3.4E-4

++The data presented for Prairie Island are based on Table 4.7-1 of the IPE submittal, which include the contribution from induced SGTR (30% of CDF). According to the licensee's responses to the RAI, ISGTR is effectively precluded by a procedure change recommended by the Westinghouse Owners Group and implemented at Prairie Island. The probability of bypass failure would decrease, and intact containment increase, by 30% if ISGTR is eliminated.

* Included in Early Failure, approximately 0.02%

** Included in Early Failure, approximately 0.5%

As shown in the above table, the conditional probability of containment bypass for Prairie Island is 44.7% of total CDF. Most of it is from induced steam generator tube rupture (30% of total CDF). Excluding ISGTR, containment bypass is primarily from SGTR as an initiating event (13.2%). The contribution from ISLOCA is small (0.5%), but it results in the highest releases.

The conditional probability of early containment failure for Prairie Island is about 0.8% of total CDF[3]. It is about equally contributed by internal flooding (28% of early failure), small LOCA (26%), transient (20%), and medium/large LOCA (20%) accident classes. The contribution from SBO sequences is only about 6%. The smaller contribution from SBO sequences is partly due to the low CDF of SBO sequences (about 6% of total CDF). Early containment failure is primarily due to hydrogen burn with vessel breach at high pressure (74% of early failure CDF). The remaining 26% of early failure is also due to hydrogen burn, but with no vessel failure, primarily from medium/large LOCA sequences.

The conditional probability of late containment failure for Prairie Island is 22.6% of total CDF. More than half of this is from small LOCA sequences (59%), with most of the remaining coming from transient sequences (38%). On a conditional basis, 58% of small LOCA sequences result in late containment failure and 39% of transient sequences results in late containment failure. The contribution from SBO sequences to late containment failure (2% to total late failure CDF and 7% conditional probability) is relatively low in comparison with that from small LOCA or transient sequences. The low failure probability for SBO is probably due to ac recovery. The time allows for ac recovery is long for late containment failure.

For Prairie Island, late containment failure is primarily due to overtemperature/overpressure failure caused by lack of cooling to the debris dispersed to outside of the reactor cavity (78% of late failure CDF). Because a significant amount of core debris is dispersed to outside the reactor cavity only in high pressure vessel failure cases, the conditional probability of late containment failure is low for accident classes that involve low pressure vessel failure or no vessel failure (e.g., medium/large LOCA and the late core melt accident classes). Besides the above late failure mechanism, another important late failure mechanism is the loss of decay heat removal (22% of late failure). The contribution from basemat melt-through is small because core debris is assumed to be coolable if RWST water is injected to the containment.

2.4.2.3 Characterization of Containment Performance

As shown in Table 10, for Prairie Island Nuclear Generating Station, the core damage frequency (CDF) is lower than that obtained in NUREG-1150 for Zion but comparable to that obtained in NUREG-1150 for Surry. Although the conditional probability of containment bypass obtained in the Prairie Island IPE is significantly greater than those obtained in NUREG-1150 for Surry and Zion, it is similar to that for Surry if the probability of ISGTR is eliminated. For the other failure modes, the conditional probability of late containment failure for Prairie Island is similar to that for Zion and the probability of early containment failure is between those obtained in NUREG-1150 for Surry and Zion. The containment failure profile obtained in the Prairie Island IPE is in general consistent with those obtained in NUREG-1150.

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[3] The fractional contributions discussed below are obtained from the C-Matrix provided in the licensee's response to RAI (Level 2 Question 7). There are minor differences between the some values presented in the IPE submittal and those obtained from the C-Matrix (probably due to truncation or round-off errors).

The C-Matrix, which shows the conditional probabilities of CET end states (or containment failure modes) for the accident classes (or PDSs), is provided in the licensee's response to RAI (Level 2 Question 7). The C-Matrix provided in the response does not include ISGTR failure. According to the response, the procedure to restart the RCPs, which has a significant effect on ISGTR, is changed based on the recommendations of the Westinghouse Owners Group. This change would in effect eliminate the ISGTR challenge.

2.4.2.4 Impact on Equipment Behavior

The effects of harsh environmental condition on the operation of containment sprays and containment fan coolers are not discussed in the CET quantification of the IPE submittal but are discussed in the licensee's response to RAI (Level 2 Question 9). In the response, the potential adverse effect of containment pressure and temperature, radiation, and aerosol and debris on the operation of containment spray and fan coolers are discussed. Although the effect of aerosols and debris on the operation of the above containment systems are not considered in CET quantification, the sensitivity studies performed in the IPE can be used to address the effect of the loss of these systems on containment failure.

2.4.2.5 Uncertainties and Sensitivity Analysis

Sensitivity studies are discussed in Section 4.8 of the IPE submittal. Two types of sensitivity studies were performed in the Prairie 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, performed in the IPE to establish the sensitivity of the Level 2 analysis to uncertainties in the physical modeling of containment response and the source term.

The sensitivity studies of the first type performed in the IPE include:

1. Retention of the debris in the reactor vessel by submerging the lower vessel head (i.e., ex-vessel cooling for in-vessel recovery),
2. Depressurization of the reactor by hot leg creep rupture,
3. Debris coolability in the reactor cavity, and
4. Cooling of the debris relocated to the upper parts of the containment following HPME.

Results of the sensitivity studies show little effect on containment failure probabilities by the assumption used for in-vessel recovery by ex-vessel cooling and the assumption used for hot leg creep rupture (Items 1 and 2 above). On the other hand, the assumption on the coolability of the debris in the reactor cavity has a significant effect on late containment failure. Assuming that the debris is not coolability and long-term containment failure occurs once the debris is discharged from the reactor vessel (i.e., the vessel fails), the conditional probability of late containment failure would increase from 21% for the base case to 63%. Correspondingly, the conditional probability of no containment failure is decreased from 65% to 23%. Since in this sensitivity case containment failure is assured if the vessel fails, the cases for no containment failure (23%) are largely made up of sequences in which the event is terminated by ex-vessel cooling (i.e., with the lower vessel head submerged).

For the remaining sensitivity case, containment failure by the relocated debris is assumed to occur even with the operation of containment sprays. Results obtained for this sensitivity case are similar to those obtained for the sensitivity case on debris coolability in the reactor cavity (sensitivity case 3 above) - The conditional probability for late failure increases from 21% for the base case to 63% for the sensitivity case. The similarity

of these two sensitivity cases reflects the fact that the majority of core damage events occur at high pressure with a significant amount of debris relocated to the upper compartment.

Deterministic analyses of the following categories are performed in the IPE to evaluate the uncertainties in physical modeling of containment response and the source term:

1. Core melt progression and in-vessel hydrogen generation,
2. Natural circulation, induced ruptures of the primary system, and RCS pressure at vessel failure,
3. Fission product release and revaporization,
4. Ex-vessel debris coolability,
5. Energetic events in containment, and
6. Containment failure modes.

MAAP code is used for the sensitivity studies of the above issues, which are identified in the Prairie Island IPE by the recommendations made in the EPRI Guidance Document for performing sensitivity studies with MAAP 3.0B, the augmentations to these recommendations provided in the NRC sponsored MAAP 3.0B code evaluation, and specific areas deemed importance for Prairie Island.

The sensitivity studies provided in the Prairie Island IPE seems to have addressed the issues of significant uncertainties in the IPE analysis.

2.5 Evaluation of Decay Heat Removal and CPI

2.5.1 Evaluation of Decay Heat Removal

2.5.1.1 Examination of DHR

The IPE addresses decay heat removal (DHR). Several methods of DHR are discussed, including secondary cooldown and depressurization (using either AFW or main feedwater providing the steam generator makeup), feed and bleed (i.e., utilizing the SI pumps and pressurizer PORV), safety injection and recirculation cooling (as provided by the SI and RHR systems), and shutdown cooling (by the RHR operation) . In addition containment cooling is mentioned.

The CDF contributions from each of the individual DHR methods were not estimated either in the submittal or in the RAI response.

2.5.1.2 Diverse Means of DHR

The submittal provides a fairly detailed description of each of the diverse DHR capabilities at the Prairie Island plant. The description includes a reiteration of several specific (maybe "unique") features of the systems involved in DHR and major modeling assumptions, a discussion of the effects of initiating events on the systems' unavailabilities, and a presentation of the important hardware failures and operator errors contributing to these unavailabilities.

The specific features that directly impact the ability of the systems to provide DHR are described in Section 1.2 ("Key Features"). Therefore, they will not be discussed here again.

It is noteworthy that the steam relief part of the secondary side heat removal was not modeled for the Prairie Island IPE. The licensee's reason for it is the large diversity of means of steam removal. ("Following a reactor trip, steam is relieved to the condenser through a single air operated relief valve or to the atmosphere through four air operated valves. If the MSIVs should fail closed, steam relief is possible through an air operated PORV for each steam generator or through five safety relief valves on each steam generator, all of which are upstream of the MSIVs. In the event of loss of air, DC control power or instrument power, steam relief is assured through the five safety valves for each steam generator as they are not dependent on any support systems.") Therefore, it was assumed, that the unavailability of secondary cooling is primarily determined through the probability for a loss of makeup capability to the SGs.

Table 11 summarizes the unavailability values of the systems involved in the DHR for selected initiating events or initiating event groups. The data markedly demonstrate the strong dependency of the systems' unavailabilities on the type and nature of the initiating events. This dependency is illustrated by the following examples:

The unavailability of feed & bleed is relatively low for normal transients, since its operation is principally dependent on operator action to initiate the process. For initiators, like loss of DC train A or B, the instrument air containment isolation valves fail closed cutting off IA to the pressurizer PORVs, failing feed & bleed.

The unavailability of the AFW system can vary for the spectrum of initiators. It has a rather low value for transients (other than LOOPs) because the AFW support systems include only AC and DC power with cooling water providing a redundant suction source in the event the condensate storage tanks are depleted. The increase of unavailability for LOOP events and for SBO reflects the additional dependence on the DGs. In the event of an SBO, the TDAFW pump is the only means for feedwater addition to the SGs. This pump is not dependent on AC or DC power, as the steam admission valve to the pump fails open on loss of DC power.

The submittal explicitly provides the important hardware and major human error contributions to the unavailability of the various DHR systems (expressed in %, but understood as "importance"). These offer valuable insights about the reliability of the systems, therefore they are reproduced below.

Auxiliary Feedwater System: Random failure of Unit 1 TDAFW pump to run (79%), similar failure for Unit 1 MDAFW pump (27%), misalignment of the MDAFW pump train after test & maintenance (24%), and MDAFW pump motors fail to start due to common cause (22%).

The operator error "Failure to crosstie the MDAFW pump from Unit 2" contributes 51% to the system's unavailability.

The present data on the unavailability of the AFW system reflect the recent improvements implemented on the system (see the report, NSPNAD 8606P Rev.0).

Table 11 Effects of Selected Initiating Events on the Unavailability of Systems Involved in DHR

Initiating Event/ Event Group	Secondary Side Heat Removal*		SI		RHR			Feed & Bleed
	AFW	MFW**	Inj.	Recirc.	Inj.	Recirc.	Shut-down Cooling	
Large LOCA					1.3E-04	1.2E-02		
Medium LOCA			1.9E-03	5.9E-03	1.3E-04	1.2E-02		
Small LOCA			1.9E-03	5.9E-03			1.7E-02	
SGTR			1.9E-03	5.9E-03			1.7E-02	
LOOP (SBO)	8.1E-04 (3.6E-02)	1.0		1.6E-02				5.1E-02+
Loss of CC				5.8E-01*				5.2E-01*
Loss of CL		1.0		5.8E-01*				5.2E-01*
Loss of IA		5.8E-01*						5.8E-01
Loss of DC Transient A		1.0						1.0
Loss of DC Transient B								1.0
Transients	8.3E-05			6.5E-03				
Transients with "s" signal		3.9E-02						4.7E-02+
Transients w/o "s" signal		6.6E-03						4.1E-02+

*Steam relief was not modeled because of the several diverse means of steam removal.
**MFW is not a safeguards system.
*Includes recovery of initiating event.
+Value depends on the operator action to start F&B and the unavailability of DGs.

Main Feedwater System: Bus 110 unit cooler unavailable due to maintenance (15%), control room chillers fail to run due to common cause (4%), control room chilled water pumps fail to run due to common cause (4%). All of these hardware failures cause eventual loss of DC power via loss of room cooling to the safeguards 480 V bus rooms. Namely, if room cooling is lost the transformers in the rooms will heat up and eventually fail causing loss of all loads supplied by the affected buses. In this way the battery chargers will be lost causing DC power to fail after the batteries have depleted. (This condition has been changed, as mentioned previously in this TER.) Important operator action is the restoration of MFW for events in which it lost as a result of the initiating event but is otherwise available (60%).

Feed & Bleed: Hardware failures are insignificant contributors to the unavailability of feed and bleed. Almost 95% of the unavailability is from human error to initiate the process. (The process initiation is different in the presence or absence of an S signal: if there is no S signal, the operator must manually start the SI pumps and open the pressurizer PORV, while if an S signal has been generated, the operator has to verify only whether the SI pump is running and then open the pressurizer PORV.)

SI Injection: Boric Acid Supply Tank (BAST) suction valves fail to open due to common cause (25%), RWST suction valves fail to open due to common cause (25%), control room chilled water pumps fail to start due to

common cause (19%), both SI pumps fail to start due to common cause (9%). The SI pumps draw suction off the BASTs during the first few minutes of the injection phase of an accident, and then switch to the RWST when the Lo-Lo level alarm is reached on the BASTs. Chilled water failure would cause overheating of the Unit 1 safeguards 480 V bus rooms and failing of all loads supplied by the safeguards buses. This would mean for the SI system, that the suction valves from the BAST and the RWST would fail as they are the only valves required to change state for successful system operation. The unavailability calculation included recovery actions, such as: local recovery of the suction valves, recovery of 480 V room cooling. The submittal, however did not provide any quantitative information about their total unavailability contribution.

SI Recirculation: Both CC supply valves to the RHR heat exchangers fail to open due to common cause (7%), both RHR to SI crossover supply valves fail to open due to common cause (7%), control room chilled water pumps fail to start due to common cause (4%), and both control room chillers fail to start due to common cause (4%). The single dominant operator failure for SI recirculation is to initiate the recirculation (41%), since the lineup for the SI recirculation cannot be performed from the control room (the breakers for the RHR to SI crossover valves are locked in the off position and they have to be unlocked locally).

RHR Injection: Both RHR pumps fail to start (57%) and fail to run (29%) due to common cause. RWST rupture (5%).

RHR Recirculation: Both CC supply valves to the RHR heat exchangers fail to open due to common cause (4%), both control room chilled water pumps fail to start (3%) and chillers fail to run (2%) due to common cause. The dominant operator failure (72%) is to initiate the low head recirculation. (Switchover to low head recirculation can be performed from the control room; the limiting factor is the time available to perform the action before the RWST would deplete. During large LOCA the RWST level is expected to fall rapidly.)

RHR Shutdown Cooling (RHR SDC): RHR Loop B return valve fails to open (28%), failure of Train A control room chilled water pump to start (19%) and to run (13%), and failure of the chiller in the same train to run (13%). Failures associated with the chilled water system cause the known heatup of the 480 V safeguards bus rooms, resulting in the loss of the motor control center that powers the single Loop B return valve, failing RHR SDC.

Finally a remark has to be made on the role of the containment fan cooler units (FCUs) and spray (CS) systems in the process of DHR: In the Prairie Island Event Trees success or failure of recirculation is asked before asking the status of the top event "containment " (i.e., the FCU and CS). If recirculation fails, it is assumed that core damage occurs. Credit is not given for the FCUs to remove decay heat from containment and condense the water to return it to the containment sump. (Thus, according to this assumption, failure of the RHR heat exchanger results in failure of the recirculation even though the RHR pumps could recirculate the water through containment where the heat could be removed by the FCUs.)

2.5.1.3 Conclusion On the Analysis of DHR

The NRC defines two requirements in NUREG-1289 that have to be met by any system which performs DHR. These are:

1. Maintain sufficient water inventory in the RCS to ensure adequate cooling of the fuel.
2. Provide the means for transferring heat from the RCS to an ultimate heat sink.

In the IPE there are no core damage sequence that do not involve loss of either one or both of the two requirements. This fact lead the licensee to consider the loss of DHR to be synonymous with core damage. Since the overall CDF was found to be acceptably low (5 E-05/yr) and it was shown, that there are several redundant and diverse means for DHR, (i.e., several of the DHR systems and operator actions would have to fail in combination to have a serious negative impact on the DHR capability), the licensee considered that it has fulfilled the "Shutdown Decay Heat Removal Requirements" of "Unresolved Safety Issue A-45," (USI A-45).

2.5.2 Other GSIs/USIs Addressed in the Submittal

No GSIs or USIs, other than USI A-45 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 10). Hydrogen sources, the condition for hydrogen detonation, and the load for hydrogen deflagration are discussed in the response. According to the response, hydrogen detonation is highly unlikely to occur with the Prairie Island containment geometry and hydrogen concentration. The loading condition generated by hydrogen deflagration has been pessimistically treated in the IPE and found not likely to cause containment failure.

2.6 Vulnerabilities and Plant Improvements

The criteria used in the Prairie Island IPE to determine whether any vulnerability existed at the plant were:

1. Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRAs?

2. Is there adequate assurance of no undue risk to public health and safety?

The licensee states that neither of the above criteria lead to the identification of potential vulnerabilities for the plant. The IPE process demonstrated that; the accident classes contributing to the CDF are comparable with those calculated in PRAs of similar nuclear plants (indeed the comparison made in Table 2.4-1 of the submittal with Surry, Kewaunee, Point Beach supports this viewpoint), and the overall CDF itself is at an acceptably low level of 5E-05/yr. Therefore the licensee believes that there is adequate assurance of no undue risk to public health and safety.

The licensee states that while no vulnerability exists, as a result of the IPE, recommendations have been generated for plant improvements. These recommendations are partly implemented and partly only under consideration but by no means represent any definitive "NRC commitments". The recommendations focus on plant improvements in three areas: procedural/administrative, structural and training enhancements.

Depending on further evaluation of potential benefits and practicality, the licensee expects a significant decrease in the overall CDF (a decrease of 1E-05/yr or greater). In particular, the risk contribution from internal flooding is anticipated to be reduced from 1E-05/yr to approximately 1E-08/yr.

A summary of the Level 1 related recommendations is provided below:

Procedural/Administrative Enhancements

1. The plant already proceduralized the process establishing crosstie from station air to instrument air in C34 AOP1, Rev.0.

   a) If the crosstie could be established within one hour after a CL Loop A break, feed & bleed or main feedwater cooling could be restored and core melt could be prevented.(The station air compressors are cooled from Loop B cooling water and are not affected by a break in the other loop.)

   b) The new procedure prescribes also that the station air crosstie should be used when an IA compressor is in maintenance.

Present (November 28, 1995) disposition of the recommendation: a.) C34 AOP 1, Rev. 4 incorporates this action (Step 2.4.6); b) C34, Rev 12 incorporates this recommendation.

2. The procedure C35 AOP1, Rev.2, "Loss of CL Water Header A or B" should be revised such that the crosstie between CL Loop A and B could be used. Two valves, one manual and an AOV have to be opened during 20 minutes to supply cooling water to the MFW pumps' lube oil coolers. The MFW pumps can conservatively operate w/o cooling water for about 20 minutes before possible pump damage.

Present disposition of the recommendation: See the next disposition.

Structural Enhancements

1. Constrain the impact of AFW pump room flooding by some simple measures. Evaluations are underway to determine the best long term solution. In the interim the following measures are suggested; modify the side doors to promote water flow out of the room, or close the fire door between the two halves of the room and render the door to be "water tight".

Present disposition of the recommendation: The CL header piping was completely replaced with a piping of 33% thicker wall during the November, 1992 dual-unit outage. The internal surface of the new pipe is coated with an epoxy coating to inhibit microbiologically induced corrosion (MIC). Also piping failure would be noticed by any personnel who periodically walk through these rooms.

Enhancement of Training

1. Explain the importance of the feed & bleed process. Put an emphasis on the operator actions that are necessary for success. It is expected that the training will result in marked reduction of the contribution of accident class THE to the overall CDF.

2. Explain the importance of a crosstie between the MDAFW pumps. Emphasize the operator actions that are required for success. It is expected that due to the training a significant reduction will occur also in the THE's contribution to the overall CDF.

3. Explain the importance of switchover to high and low head recirculation. Emphasize the operator activities that are required to success. The training may reduce significantly the CDF contribution of the accident class SLL.

4. Explain the importance of RCS cooldown and depressurization to terminate SI before ruptured SG overfill. Emphasize the operator actions that are required for success. It is expected that the training will result in reduction of the CDF contribution of accident class GLH.

Disposition of the above recommendations:

a) Letter 2.21.94, M. Wadley to D. Reynolds, asking to take the necessary actions to ensure the operators receive periodic training on the IPE recommended training actions. The letter identifies the actions and a suggested frequency for giving training on them.

b) Request for Training 94-25 from J. Sorensen. Requal./NLO training on IPE and bases. Training completed during cycle 94-09.

c) Course Outline for Simulator Continuing Training: P9160S, Rev.4. Records of the IPE recommended training items at the frequency requested in the Wadley letter.

d) NLO training Program P8400, Rev.9: Outplant actions required to successfully establish low head recirculation and to cross connect the MDAFWP to the opposite unit are JPMs (SI-3 and AF-7), required biennially.

e) Lesson Plan P8161L-003, Rev. 1, Introduction to Accident Analysis for license candidates: In addition to USAR accident analysis topics, students are trained in how PRA techniques are used to determine risk, and on the results and uses of the PI IPE in the operation and maintenance of the plant.

The following two recommendations are related to the back-end:

1. Revise the EOP that requires the restart of the RCPs under ICC condition. It is recommended in the IPE submittal that the operator checks for adequate steam generator level before attempting to start the RCP. This recommendation is intended to reduce the probability of ISGTR.

2. Secure open the in-core instrument tube hatches for both units to allow water to flow into the reactor cavity to provide cooling to the lower vessel head (i.e., ex-vessel cooling) and improve debris coolability in the reactor cavity.

3 CONTRACTOR OBSERVATIONS AND CONCLUSIONS

Strengths of the Level 1 analysis of the IPE are as follows: Thorough analysis of initiating events and their impact for both units of the plant, descriptions of the plant responses, presentation of the results of supporting MAAP calculations (Section 7 of the submittal), reasonable failure data and common cause factors, usage of plant specific data whenever possible to support the quantification of initiating events and system unavailabilities, and an importance analysis on major variables impacting core damage. The effort seems to have been evenly distributed across the various areas of the analysis.

The IPE Level 1 analysis has no fundamental weaknesses. Its "leading" minor weakness is the lack of detail in the individual accident sequence descriptions. The text of the submittal everywhere indicated that a detailed analysis was done but only highlights were reported.

The IPE determined that an internal flooding sequence is the primary contributor to core damage. Its initiator is a cooling water line break inside either the Unit 1 or Unit 2 side of the Turbine Building AFW/IA compressor room, which causes failure of these two systems for both units. There is a low potential for this pipe rupture (only a small section of piping is involved); however, this initiating event represents a potentially important location dependency for several systems at the plant, since the AFW pumps (secondary heat removal) and the IA compressors (in feed & bleed support for the pressurizer PORV) are located in the same room.

As was noted previously, several recommendations for plant improvements have been made as a result of insights obtained from the IPE, particularly to reduce the CDF contribution of the above mentioned flood accident. The CDF impact of these improvements is expected to be a CDF decrease on the order of 1E-05/yr or greater.

The HRA review of the IPE submittal did not identify any significant problems or errors. A viable approach was used in performing the HRA and nothing in the licensee's submittal indicated that it failed to meet the intent of Generic Letter 88-20 in regards to the HRA. Important elements pertinent to this determination include the following:

1) The submittal indicated that utility personnel were involved in the HRA and the procedure reviews, discussions with operations and training staff, and walkdowns of operator actions represent a viable process for confirming that the HRA portions of the IPE represent the as-built-as operated plant.

2) The analysis of pre-initiator human actions focused on restoration faults. The HEPs assigned to the modeled restoration faults and the approach for computing the component unavailabilities appeared reasonable. Dependencies between restoration errors were not addressed because it was argued that plant practices regarding maintenance of separate trains assured the independence of restoration faults. Miscalibration errors were "treated through the inclusion of common cause failure modeling for the sensors or instruments themselves." The licensee's treatment of miscalibration events may have precluded identification of important pre-initiator events and is therefore a weakness of the HRA.

3) Post-initiator human actions modeled included both response-type and recovery-type actions. Although the documentation for the screening analysis performed on post-initiators was minimal, it appeared that the screening analysis was relatively more "plant-specific" and detailed than many of those performed for other IPEs. In addition, the licensee stated that if more than one operator action occurred in the same cutset, "either independence of the human actions was confirmed, or a change was made to correctly model dependence between human errors." The licensee's response to an additional RAI regarding treatment of dependencies confirmed that they were appropriately addressed. Moreover, the detailed quantification of important post-initiator operator actions and the quantification of recovery actions appeared sound.

4) Plant-specific performance shaping factors (PSFs) and event timing, were appropriately considered.

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

The IPE uses five small containment phenomenological event tree (CET) with from 5 to 8 top events for Level 2 analysis. The quantification of the CET in the Prairie Island IPE is based on plant-specific phenomenological evaluations, which include modeling and bounding calculations (based upon experimental data), consideration of phenomenological uncertainties, and MAAP calculations.

The interface between the Level 1 and Level 2 analyses is accomplished by the development of a set of 14 accident classes. The Level 1 core damage sequences are grouped in the accident classes based on accident initiators, the time of core damage, and the RCS pressure at the time of core damage. The availability of containment systems is not included in the definition of accident classes. Containment system fault trees (for containment spray injection, containment spray recirculation, and containment fan coil units) were quantified in the Prairie Island IPE using linked fault tree models. The containment systems fault tree cutsets were input to the CET branches as necessary to support the CET quantification. The definition of the ACs for 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 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 Prairie Island Nuclear Generating 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 in-core instrument tube hatches for both units be secured open to allow water to flow into the reactor cavity to provide cooling to the lower vessel head (i.e., ex-vessel cooling) and improve debris coolability in the reactor cavity.

- The steel shell containment of Prairie Island may be vulnerable to direct attack by dispersed core debris. The access hatches to the instrument tunnel are in an open area on the basement level of the containment, and for both of the Prairie Island units one of the two hatches faces toward the steel

containment, about 30 ft away, with a largely unobstructed path in between. Although a scoping study performed in the IPE shows that the temperature generated by the debris adhering to the steel wall is insufficient to melt the steel and breach the containment, details of the scoping study are not provided in the submittal and the potential effect of corium attack on reducing containment pressure capability is not discussed.

- The IPE identified the potential of ISGTR due to the restart the RCPs upon an inadequate core cooling (ICC) condition. Based on the IPE results, it is recommended in the IPE submittal that the EOP that requires the restart of the RCPs under ICC condition be modified. It is recommended in the IPE submittal that the operator checks for adequate steam generator level before attempting to start the RCP. This recommendation is intended to reduce or eliminate the probability of ISGTR.

- The containment analyses indicate that there is a 68% conditional probability of containment failure if ISGTR due to restart of the RCP is included. The conditional probability of containment bypass is about 45% of which 30% is from ISGTR, the conditional probability of early containment failure is 0.8%, the conditional probability of isolation failure is about 0.02%, and the conditional probability of late containment failure is 23%. The conditional probability of containment bypass decreases by 30% if ISGTR is assumed not to occur.

The licensee has addressed the recommendations of the CPI program.

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4 REFERENCES

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