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TECHNICAL EVALUATION REPORT OF THE IPE SUBMITTAL AND RAI RESPONSES FOR THE BIG ROCK POINT NUCLEAR POWER 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 Big Rock Point (BRP) Nuclear Power Plant. The primary purpose of the review is to ascertain whether or not, and to what extent, the IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 and achieves the four IPE sub-objectives. The review utilized both the information provided in the IPE submittal and additional information provided by the licensee, the Consumers Power Company (CPC), in the response (RAI Responses) to an NRC request for additional information (RAI).

E.1 Plant Characterization

The BRP Nuclear Power Plant is a 75 MWe, 240 MWth General Electric boiling water reactor (BWR) of BWR-1 design. The reactor coolant system (RCS) consists of the reactor vessel, the main feedwater and the steam system, the steam drum, the emergency condenser, outside pump driven recirculation loops and interconnected piping. The plant is operated by Consumers Power Company (CPC), and started commercial operation in December 1962. There are no other operating units on site.

Design features at BRP that impact the core damage frequency (CDF) relative to other BWRs are as follows:

1) Large primary water inventory relative to core thermal power and decay heat levels. Over 35,000 lbm of water covers the core in the reactor and the steam drum following a reactor trip. Therefore, it would take 2 hours to deplete inventory to the top of the core even if no decay heat removal systems were to function. This feature decreases the core damage frequency relative to other BWRs.

2) Emergency condenser (EC) for high pressure core cooling and makeup. This system is similar to the isolation condenser found at some other older BWRs. The system enables passive cooling of the reactor, without reliance on ac power (diesel driven pumps can be used for shell side makeup). The time scales for success of various steps in EC operation are relatively long: with no makeup supply for shell cooling available, the emergency condenser operation can prevent safety valve lifting (setpoint at 1535 psig) for a period of 6.5 hours. With makeup available, Big Rock Point has sufficient dc capacity to cope with station blackout conditions for over a week. In case of ATWS, the emergency condenser can remove full power for 30 min before depletion of shell side inventory, thus giving the operators additional time to respond.

   This feature decreases CDF vulnerability relative to most other BWRs.

3) The firewater system, consisting of one diesel driven and one electric pump, can be used as part of the ECCS. This system can be used for low pressure injection (i.e., core spray), for cooling of the post incident (i.e., recirculation) heat exchangers, or for "fill the ball" (used when recirculation is not available). It can also be used for emergency condenser secondary side makeup (as backup to demineralized water system), or to provide inventory for the main condenser. There is also an additional, and portable, diesel driven pump, which can be used for emergency condenser makeup. This feature tends to decrease the CDF.

4) Lake Michigan can be used as the ultimate heat sink. The fire system takes suction from Lake Michigan. This feature decreases the CDF.

5) In case recirculation is unavailable for continued core cooling, "fill the ball" can be used. This involves continuing in the injection mode, thus almost filling the spherical containment with water. The post-incident (or recirculation) system is initiated when the water level reaches the 587 ft elevation inside the containment, in order to preserve containment integrity. The maximum permissible containment water level is 596 ft, based on the design pressure (about midplane). Depending on how many ECCS pumps are operating, this level will be reached in between 6.3 and 21 hours after the initiator. Should the post incident system fail, fill the ball is used (i.e., injection is continued until most of the 130 ft diameter sphere is filled with water), as calculations show that the containment can really withstand fill up to about 620-636 ft level (depending on internal air pressure), at which point injection is terminated and passive cooling via natural circulation and air cooling of the containment steel shell takes over. However, although the operators are trained in this procedure, no credit is given for fill the ball operation in the IPE. If credited, this feature would decrease the CDF.

6) The emergency ac power consists of one 200 kW emergency diesel generator, and one 250 kW, standby diesel generator. There is only one safety bus, and the diesels are sized for just one CRD pump and/or the electric fire pump (and/or the demineralized water system for emergency condenser makeup). The absence of diesel capacity for any other pumps would be detrimental, compared to other BWRs, were it not for the fact that the plant relies extensively on passive features. The core cooling function can be accomplished without ac power. DC power is supplied by two battery banks, i.e., the normal and the alternate shutdown battery. The alternate shutdown battery supplies the post-initiator loads of interest, and is sized such that a blackout can be survived for about a week, a positive plant feature. The emergency power system requires no support function.

7) CRD pumps cannot be used in conjunction with safety valve cycling or actuation of the reactor depressurization system due to high temperature in the CRD pump room, a negative plant feature. In general, this plant seems to be more vulnerable to environmental conditions (some of the other systems vulnerable to harsh conditions are the reactor cooling water system, emergency condenser outlet valves, primary core spray valves, the reactor pressure and level instrumentation, as well as some operator actions).

8) The instrument air system has three air compressors, with one being sufficient for system success. Apparently domestic water can be used for backup cooling of air compressors. These are positive features. (A fourth compressor has recently been added, but is not credited in the analysis).

9) The plant has a recently improved "100% load rejection capability", a positive feature, however, not entirely proven in practice (before the recent improvement this feature never actually worked when called upon). (The plant also has a 100% turbine bypass capacity, however this feature needs additional operator actions for success).

10) No high flow rate high pressure ECCS pumps, a negative feature, except in some ATWS sequences where this prevents containment failure.

11) A fast acting, passive, manually initiated liquid poison system in case of ATWS, a positive feature.

12) A single two train low pressure ECCS for LOCA evolutions, a negative feature.

13) A portion of primary system piping which is located below core midplane, a negative feature.

14) No sign of the classic symptoms of IGSCC (inter granular stress corrosion cracking) found at other BWRs, a positive feature.

15) The large dry containment which effectively decouples containment considerations from the Level 1 analysis, compared to other BWRs, a positive feature.

Other features are described in Section 1.2.

The Big Rock Point (BRP) Plant utilizes a spherical steel vessel for a large dry containment. The plant is designed such that operating personnel may enter the sphere and remain inside as necessary during normal operation, shutdown, and refueling. In comparison with other plants that use large dry containments (PWRs), the containment volume to thermal power ratio for BRP is significantly (about four times) higher.

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

- The only BWR plant that uses a large dry containment (a spherical steel vessel).
- A containment that can be accessible during power operation. The plant is designed so that operating personnel may enter the sphere and remain inside as necessary during normal operation. The potential for a single access door to be open while personnel are entering or leaving the containment thus exists. Interlocks on the doors prevent simultaneous opening of both equipment lock doors or both personnel doors.
- The small core and large containment volume. The containment volume to core thermal power ratio is about 5 times that of PWRs with large dry containment. The large containment volume reduces the challenges to containment integrity from containment pressurization mechanisms. It also provides significant passive heat removal capability through the containment shell and other passive heat sinks.
- The capability to flood the containment. Procedures are in place at BRP to direct the operators to fill the containment vessel with water (called "fill-the-ball" in the IPE submittal). This provides cooling to the core debris in-vessel such that vessel failure may be avoided, or provides cooling and scrubbing of the debris ex-vessel if vessel failure is not prevented.
- A sump beneath the reactor vessel that has the volume to hold the entire core debris. The sump in the CRD room beneath the reactor vessel has a depth of 3 feet and a volume of 126 cubic feet. This sump may hold the entire BRP core after vessel breach (with a total depth of 1.6 feet, or 50 cm).

E.2 Licensee's IPE Process

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

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

The IPE Level 1 model (initiated in 1992) is an update of an earlier BRP Level 1 PRA, which was submitted in 1981 and reviewed by the NRC. Model updates reflect the plant modifications and data since 1981. The freeze date for the analysis was late 1992.

It appears the licensee intends to maintain a "living" PRA, and reference is made to future use of the BRP IPE models in ongoing risk management activities.

Licensee personnel were involved in all aspects of the analysis. In-plant expertise was already existent due to the previous BRP PRA study. Specialized help for aspects of Level 1 analysis and review were provided by Gabor Kenton & Associates and Tenera, and by an independent consultant.

The reviews performed for the IPE included both independent in-house reviews and an external review. The internal review was extensive and consisted of work by managers and key personnel from key organizations of the utility. External peer review was performed by the above named consultants. Some comments and their disposition from the external review are documented.

The submittal states that the Big Rock Point (BRP) HRA was performed by a combination of plant personnel and contractors from Tenera. It appears that plant staff had the lead in identifying the human actions to be modeled and for collecting relevant information regarding those events. Plant staff shared responsibility for quantifying pre-initiator events and assisted the contractors in quantifying the post-initiator events. Procedure reviews, discussions with operations and training staff, observations of simulator exercises, review of the "control room design review", and walkdowns of important operator actions, including local actions, helped assure that the IPE HRA represented the as-built, as-operated plant. Contractors (not named) and training, operations and management personnel performed a review of the HRA. 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 procedure related enhancements were discussed in the licensee's response to the NRC's RAI.

E.3 IPE Analysis

E.3.1 Front-End Analysis

The methodology chosen for the front-end analysis was a Level 1 PRA using the small event tree-large fault tree methodology. The computer code used for modeling and quantification was IRRAS.

The IPE quantified the following initiating event categories: 8 LOCAs, 7 steam line breaks, 9 transients, 7 ATWSs, 2 special initiators (flooding and ISLOCA) and 2 manual shutdown initiators. The IPE developed 37 event trees to model the plant response to these initiating events. The flooding analysis was a relatively comprehensive analysis, but with pipe failures the chief mechanism of causing the flood.

Success criteria were based on existing information (e.g., BWROG) supplemented by calculations, as needed.

For reactor depressurization, a conservative criterion of 3/4 RDS valves opening was assumed, whereas in reality in all but two sequences 1/4 valves is sufficient.

Impact of harsh environments on systems is considered. Specifically, CRD pumps, core sprays, emergency condenser and some instrumentation (reactor level) are vulnerable to steam environments in some accidents. Water collection and interaction with electrical equipment due to steam condensation is also considered.

Other types of dependencies were also considered, including HVAC. The HVAC is not needed within the mission time.

RCP seal LOCA is not considered due to the design and test results of recirculation pump seals at BRP.

The data collection process period was from 1982 through 1992, except for initiators where the period 1964-1992 was used. Plant specific component failure data were used to update generic data with the use of Bayesian techniques.

BRP data are generally consistent with the NUREG/CR-4550 data. Some of the initiating event frequencies (spurious RDS, small LOCAs) seem low. The generic and Bayesian updated data for diesel pumps seem low; the licensee provided a reasonably complete discussion of such in the RAI responses, along with a sensitivity analysis to see the impact of higher values on the CDF results.

The multiple Greek letter (MGL) approach was used to characterize common cause failures. The CCF parameters used are generally consistent with the NUREG/CR-4550 recommended values. The process used to arrive at these values follows established procedures, specializing the generic occurrences to the plant specific design and configuration. A potential weakness is that CCF between the normal and alternate plant battery is not considered (due to different location and maintenance procedures); nor is the CCF between the electric and diesel driven fire pump considered (the pumps themselves have the same design).

The internal core damage frequency is 5.4E-5/yr. The flooding contributes an additional 1.1E-9/yr. The internal accident types and initiating events that contribute most to the CDF and their percent contributions are listed below in Tables E-1 and E-2. Several sensitivity and importance analyses were performed, including also Fussell-Vesely and Birnbaum importance of systems. The discussion of these subjects is very comprehensive and thorough. Calculation of Birnbaum importance for a few systems was wrong in the submittal, but this was corrected in the RAI responses.

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

Initiating Event Category	Annual Frequency	%CDF
LOCA below core	3.2E-5	59.33
LOCA above core	7.7E-6	14.30
Support system transient	5.1E-6	9.44
ATWS	3.7E-6	6.96
SLB inside containment	3.4E-6	6.30
General transient	1.2E-6	2.18
Loss of offsite power group (load rejection, loss of station power, station blackout)	7.6E-7	1.42
SLB outside containment	2.5E-8	0.05
Other (ISLOCAs, flooding)	1.3E-8	0.03

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

Initiating Event	Contribution to CDF (/yr)	%
Very small LOCA below core	1.4E-5	25.27
Small LOCA below core	1.0E-5	19.50
Medium LOCA below core	6.3E-6	11.83
Loss of Instrument Air	4.8E-6	9.03
Small LOCA above core	4.6E-6	8.64
Turbine trip ATWS	3.4E-6	6.35
Medium LOCA above core	2.5E-6	4.66
Large LOCA below core	1.5E-6	2.7[illegible]
Very small SLB inside containment	1.3E-6	2.50
Small SLB inside containment	1.1E-6	1.99
Manual shutdown	8.1E-7	1.51
Medium SLB inside containment	7.7E-7	1.44
Large LOCA above core	5.4E-7	1.00
Station Blackout	5.3E-7	0.99

E.3.2 Human Reliability Analysis

The HRA process for the Big Rock Point 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. All pre-initiator restoration errors were analyzed in detail (no screening analysis) and quantified using the ASEP HRA procedure (NUREG/CR-4772). All common cause miscalibrations were quantified in detail using a method derived from THERP (NUREG/CR-1278).

The Big Rock Point IPE acknowledges both response and recovery type post-initiator human actions. However, post-initiator actions were modeled only when clear procedural guidance (normal, abnormal, or emergency procedures) existed for the operators and repair activities were apparently not credited. To account for dependencies during the initial (screening) analysis, the submittal states that "where it was initially recognized that resulting sequences may contain multiple operator actions, the HEPs were initially set to 1.0." After the initial quantification and when quantified operator actions were first included, the nominal ASEP HRA method was applied to all post-initiator human actions. Where important sequences contained multiple operator actions, the actions were analyzed to determine the dependencies between the HEPs. The HEPs obtained using the ASEP method are known to be somewhat conservative. After the sequences were quantified with the ASEP values, operator actions identified as potentially being important were re-analyzed using the THERP methodology. All the actions re-analyzed had a Birnbaum importance greater than 1.0E-6.

While in many cases the application of THERP was reasonable, there were several events for which the quantification process did not seem appropriate. It is thought that the resulting HEPs should be considered optimistic and that the use of such values for these events is a weakness of the HRA. The problem arises through the licensee's use of HEP values from the "annunciator response model" (Table 20-13 or Table 11-13 from THERP) in situations where very limited time (less than 10 minutes) is available for the operator action. While it can be argued that the HEPs from this model are acceptable when substantial time (greater than 30 minutes) is available for the operators to determine the relevant actions and when the operators need only respond to the existence of an annunciator in the control room, the HEPs from this model do not reflect the impact of the time available on the likelihood of success. Thus, this model will underestimate HEPs for short-time frame scenarios relative to the ASEP/THERP time-reliability diagnosis model. Nevertheless, the submittal still identified the events of concern as being relatively important in terms of contribution to CDF and a sensitivity analysis indicated that substantial increases in CDF would not be expected if the events were set to fail. Therefore, potential vulnerabilities related to these events were not overlooked. Other important aspects of the post-initiator HRA analysis appeared to be conducted appropriately. Human errors were identified as important contributors in accident sequences leading to core damage and several procedure related enhancements were discussed in the licensee's response to the NRC's RAI.

E.3.3 Back-End Analysis

The Approach used for Back-End Analysis

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

Quantification of the CET and its supporting logic trees is based on the review of industry literature and plant-specific analyses using the MAAP-BRP code, a computer code developed by the Department of Energy's Advanced Reactor Severe Accident Program, in corporation with General Electric, for the GE Simplified BWR (SBWR). In general, the quantification process for the CET is systematic and traceable. The results of the CET analyses lead to an extensive number of end states, which are binned into 15 release categories. Release fractions for the release categories are calculated in the BRP IPE by MAAP-BRP. However, only release fractions for CsI are reported in the IPE submittal for the release categories. Furthermore, only ranges of release fractions are reported.

For the BRP IPE, the PDS definition scheme is reasonable. The CET is well structured and easy to understand. The CET quantification is also systematic and traceable. The IPE process is in general logical and consistent with GL 88-20. However, the reporting of only the release fractions for CsI may limit their use in a consequence analysis.

Back-End Analysis Results

The KPDSs defined in the IPE are primarily based on the type of accident sequences (or initiating events), RPV pressure, and the availability of inventory makeup. The most probable KPDS obtained in the BRP IPE involves accident sequences initiated or resulting in LOCAs for which the reactor is at low pressure with

injection failure but with inventory makeup after vessel failure (55% CDF). This is followed by a transient KPDS with low reactor pressure and with the loss of coolant inventory makeup both before and after vessel failure (9.9%), and another LOCA KPDS with the reactor at high pressure with inventory makeup available after vessel failure (9.8%).

Table E-3 shows the probabilities of containment failure modes for BRP 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	BRP IPE++	Surry NUREG-1150	Zion NUREG-1150
Early Failure	4.2	0.7	1.4
Late Failure	+++	5.9	24.0
Bypass	1.5	12.2	0.7
Isolation Failure	***	*	**
Intact	94.3	81.2	[illegible]
CDF (1/ry)	1.7E-5	4.0E-5	3.4E-4

++ The data presented for BRP are based on Figure 12.8-17 of the IPE submittal.

+++ Late containment failure is assigned a probability of 1E-4 in the CETs presented in the IPE submittal. However, results presented in Figure 12.8-17 shows a zero probability for late failure. The negligible late failure probability is due to the large containment volume and the use of a 36 hour (after vessel failure) mission time.

* Included in Early Failure, approximately 0.02%.

** Included in Early Failure, approximately 0.5%.

*** Included in Early Failure. Of the 4.2% probability of early failure about 0.5% is from leaks through penetrations.

Of the 1.5% bypass probability, only 0.06% comes from Level 1 bypass sequences (i.e., ISLOCA), while the majority comes from failure to isolated the process lines that connect to the primary system. Although LOCA is the dominant contributor to total plant CDF, the main contributors to this failure mode are transients (0.8% of CDF). This is because MSIV operator is required in some transient sequences to terminate the event, and failure to isolate the MSIV results in direct containment bypass.

The conditional probability of early containment failure for BRP is about 4.2% (of total CDF). The leading contributor to this failure mode is containment overpressure failure before vessel breach in ATWS events (3.5% CDF). This is followed by leakage through containment penetrations (0.5%), mostly from leakage through vent valves and door seals. Containment penetration leakage comes primarily from LOCA (about 80% of leakage cases) and transient sequences (about 20% of leakage cases).

Because of the large containment volume and the use of a 36 hour mission time, containment failure by the energetic events at vessel breach and long-term pressurization and thermal attack is not likely at BRP.

Source term definition in the BRP IPE is based on the fission product release time and magnitude. Although there are 15 possible release categories, the CET quantification results show only 5 release categories with non-zero frequencies (Figure 12.8-17 of the submittal). Besides a no containment failure category, all the other release categories involve early releases. The conditional probability for the no-containment-failure category is about 94%. The next release category, which contributes about 4% to total CDF, has a CsI release fraction 0.1% to 1%. It is primarily from ATWS and LOCA sequences with enclosure spray available. The

release category that has a high CsI release (i.e., greater than 20% release fraction) contributes about 1.5% to the total CDF. It is mostly from containment bypass sequences in which the enclosure spray and any water collected in the sump provide no benefit in limiting the source term severity.

The sensitivity studies performed in the BRP IPE are deterministic sensitivity studies, which were performed by varying some MAAP-BRP parameter values from their base case values and analyzing the differences in MAAP-BRP calculation results. The effects of uncertainties on CET quantification results are not addressed directly. For example, it is not clear (from the sensitivity studies presented in the IPE submittal) what is the effect of the amount of core forced out of the vessel (which is one sensitivity study item) on DCH load, and consequently, the probability of early containment failure. Although probabilistic sensitivity studies were not discussed in the IPE submittal, the uncertainties on some key containment phenomenological issues were discussed in some detail in CET quantification. For example, early containment failure due to DCH was evaluated in the IPE by the use of a decomposition event tree (DET). The top events of the DET addressed the issues of significant uncertainties for DCH. The BRP IPE seems to have addressed the issues of significant uncertainties in the IPE analysis.

E.4 Generic Issues and Containment Performance Improvements

The IPE addresses decay heat removal (DHR). CDF contributions were estimated for the following DHR methods: main condenser, feedwater, emergency condenser, reactor depressurization (RDS), condensate, core spray, post incident recirculation and containment flooding.

The following generic issues are also considered closed in the submittal:

1) USI-A43, "Containment Sump Emergency Performance"

2) Closure of the BRP Severe Accident Management Guidelines.

E.5 Vulnerabilities and Plant Improvements

The licensee defined a vulnerability as new or unusual means of reaching a situation in which core damage or containment failure would occur, or if the PRA results indicated BRP would prevent the industry from meeting published safety goals. No vulnerabilities were found.

No improvements were identified or planned. A list of improvements stemming from the 1981 PRA was provided. No SBO rule improvements resulted except for minor EOP modifications. The licensee intends to monitor and maintain the performance of components and systems with high Birnbaum importance.

E.6 Observations

Based on the level 1 review of the BRP IPE the licensee appears to have analyzed the design and operations of BRP 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 BRP; gained a quantitative understanding of the overall frequency of core damage; and considered implementing changes to the plant to help prevent and mitigate severe accidents.

Strengths of the IPE are as follows: Thorough analysis of initiating events and their impact, descriptions of the plant responses, modeling of accident scenarios, reasonable failure data and common cause factors employed and usage of plant specific data where possible to support the quantification of initiating events and component unavailabilities. Treatment of dependencies and harsh environments was thorough, as were the various sensitivity and importance analyses. The effort seems to have been evenly distributed across the various areas of the analysis. The documentation is very detailed, and there seems to have been a conscious effort to respond to the RAIs to the best of the licensee's ability.

There are some areas of concern related to the IPE but these are not expected to have a major impact on the conclusions. In the area of IE frequency, the RDS spurious opening frequency and the small LOCA frequency seem low. As far as data is concerned, data for the fire pumps seem low (corrected by a sensitivity analysis). There are questions as to the Bayesian updating algorithm used, as unreasonable values are produced for the fire pumps (the other components seem OK). The common cause failure between the two station batteries was not considered. The common cause failure between the electric and diesel driven fire pumps was not considered. Also, the system Birnbaum importance calculation algorithm seems to break down at high Birnbaum importance values (partially corrected by recalculating select systems, per RAI responses). There is a question as to why only pipe failures seem to be important in the flooding analysis. It is not clear if maintenance induced floods and spray effects were treated properly. Finally, the documentation is sometimes self-contradictory, and some parts of it seem not to have been reviewed prior to publication.

The IPE determined that LOCAs contribute about 80% to the CDF at BRP. The most important sequences have failures of the post incident system, the reactor depressurization system and/or the core spray system. The interfacing system LOCAs and containment bypass sequences show negligible contribution to the CDF. The same can be said for the flooding scenarios. The blackout contribution is small (1%), due to existence of the 100% load rejection capability, the emergency condenser, the ac independent makeup to the emergency condenser and long life of the alternate shutdown battery, as well as existence of two diesel generators (albeit with limited capability). The loss of instrument air contribution is relatively large (9%) due to its usage for emergency condenser makeup from demineralized water, feedwater flow control, feedwater pump cooling, and main condenser hotwell makeup. The ATWS contribution (7%) is governed by two opposing forces: less time than at other BWRs is available for injection of the standby liquid control system, due to nonexistence of a high pressure high volume ECCS system at BRP; however the SLCS at BRP is a fast acting one that ensures subcriticality in about 1 min after operator actuation. The 100% bypass capability is not credited as the operators have to trip the recirculation pumps in a very short time in order to avoid losing the feedwater system (even though they were able to accomplish this in training exercises). Also, the trip frequency at BRP seems to be higher than at other plants.

The BRP Level 1 risk profile does not look like that of a typical BWR, where blackout and ATWS usually dominate the core damage frequency. Here LOCAs dominate, with ATWS contributing (in the absolute sense) about the same or slightly higher CDF than most other BWRs due to the features mentioned above. The blackout contribution is much smaller than at other BWRs, as explained above. There are several reasons for the high LOCA contribution: a portion of primary piping is located below the level of the core, which leads to a more severe case of LOCAs; there is paucity of high pressure (/high flow rate) makeup systems; for larger LOCAs, makeup to the condenser hotwell is inadequate which leaves the two fire pumps as the only low pressure system available; some important systems would be disabled by the harsh environments due to LOCAs and/or steam line breaks; lack of suppression pool means that at some point recirculation must be brought into play (called the post incident system); and finally, no credit is given for fill-the-ball proceduralized action, with its passive cooling features, if recirculation fails. On the other hand, it is stated that the BRP piping is not subject to the inter-granular stress corrosion cracking (IGSCC) which plagues other

BWRs; the recirculation strainers are much less vulnerable to plugging than at other BWRs; it is also claimed that the LOCA initiating event frequencies are conservative.

No improvements are contemplated as a result of the IPE. The Fussel Vesely and Birnbaum measures will be used to identify important systems, components and operator actions and maintain their performance (i.e., their reliability and availability).

The HRA review of the Big Rock Point 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:

1) The submittal indicated that utility personnel were involved in the HRA. Procedure reviews, discussions with operations and training staff, observations of simulator exercises, review of the "control room design review", and walkdowns of important operator actions, including local actions, helped assure that the IPE HRA represented the as-built, as-operated plant.

2) The HRA process for the Big Rock Point 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. All pre-initiator restoration errors were analyzed in detail (no screening analysis) and quantified using the ASEP HRA procedure (NUREG/CR-4772). All common cause miscalibrations were quantified in detail using a method derived from THERP (NUREG/CR-1278). A reasonable and thorough analysis of pre-initiator events was performed.

3) In general, the licensee's analysis of post-initiator events was performed reasonably. A detailed "screening" was performed and important human actions were given an even more detailed analysis. However, there were several events for which the quantification process did not seem appropriate. It is thought that the resulting HEPs should be considered optimistic and that the use of such values for these events is a weakness of the HRA. The problem arises through the licensee's use of HEP values from the "annunciator response model" (Table 20-13 or Table 11-13 from THERP) in situations where very limited time (less than 10 minutes) is available for the operator action. While it can be argued that the HEPs from this model are acceptable when substantial time (greater than 30 minutes) is available for the operators to determine the relevant actions and when the operators need only respond to the existence of an annunciator in the control room, the HEPs from this model do not reflect the impact of the time available on the likelihood of success. Thus, this model will underestimate HEPs for short time frame scenarios relative to the ASEP/THERP time-reliability diagnosis model. Nevertheless, the submittal still identified the events of concern as being relatively important in terms of contribution to CDF and a sensitivity analysis indicated that substantial increases in CDF would not be expected if the events were set to fail. Therefore, even though the quantification of these events must be considered a weakness of the HRA, potential vulnerabilities related to thee events were not overlooked.

4) Plant-specific performance shaping factors (PSFs), dependencies, and event timing (with the exceptions noted in item 3 above) were appropriately considered in most instances. However, in one event the licensee *may* not have appropriately factored in the impact of potential radiation hazard on operator performance. The operator action associated with aligning the fire system for makeup to the hotwell requires a valve on top of the turbine shield to be opened. The licensee notes that "this area of the plant is not shielded from containment" and that "as a result a very short time frame is

conservatively assumed to complete this action." Presumably this statement means that the person performing this action will only be there for a short time. The licensee does not assume that there is time for an individual to put on protective clothing, but they do note that high stress was assumed for this event. Thus, the impact of radiation is apparently factored in to the HRA by assuming high stress. Additional information regarding specifics of a particular event would be needed to determine whether or not such treatment is adequate. The licensee's sensitivity analysis indicated that CDF would not increase substantially even if the action to align the fire system to the hotwell was assumed to fail.

5) 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:

1) 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.

2) The Big Rock Point Plant IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.

3) Because of the large containment volume, the probability of containment failure due to containment pressurization at and after vessel failure is not significant. The major contributor to containment failure is from containment overpressure in ATWS event (3.5% CDF) and containment bypass (1.5%) and leakage (0.5%) due to isolation failure.

4) The probability of containment leakage (0.5% CDF) may be partly due to the access of the containment during normal operation. One of the two doors of containment access locks may be open during normal operation. Leakage through door seals and vent valve leakage contribute over 90% to the total leakage probability.

5) The negligible late containment failure probability is primarily due to the large containment volume and thick concrete below the CRD room sump beneath the reactor vessel. It is also influenced by the use of a 36 hour mission time.

6) The containment analyses indicate that there is a 6% conditional probability of containment failure. The conditional probability of containment failure is about 1.5% for containment bypass, 4.2% for early containment failure, and negligible for late containment failure.

7) Only release fractions for CsI are used in the IPE for source term classification and are reported in the IPE submittal for the various source terms. Release fractions for other fission products categories are not reported in the IPE submittal.

NOMENCLATURE

AFW	Auxiliary Feed Water
ASEP	Accident Sequence Evaluation Program
BRP	Big Rock Point
BWROG	BWR Owners Group
CAC	Containment Air Cooler
CCF	Common Cause Failure
CCI	Core-Concrete Interaction
CCW	Component Cooling Water
CDB	Core Damage Bins
CDF	Core Damage Frequency
CET	Containment Event Tree
CPC	Consumer Power Company
CPI	Containment Performance Improvement
CRD	Control Rod Drive
CS	Containment Spray
CST	Condensate Storage Tank
DHR	Decay Heat Removal
EC	Emergency Condenser
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedures
FTC	Failure to Close
FTO	Failure to Open
FTR	Failure to Run
FTS	Failure to Start
GSI	Generic Safety Issue
HEP	Human Error Probability
HPI	High Pressure Injection
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning
ICC	Inadequate Core Cooling
IGSCC	Inter-granular Stress Corrosion Cracking
ISLOCA	Interfacing System LOCA
IPE	Individual Plant Examination
KPDS	Key Plant Damage States
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LPI	Low Pressure Injection
MDAFW	Motor Driven AFW
MGL	Multiple Greek Letter
PDS	Plant Damage State
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PWR	Pressurized Water Reactor
RAI	Request for Additional Information

RAV	Risk Achievement Value
RC	Release Category
RCS	Reactor Coolant System
RDS	Reactor Depressurization System
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SFAS	Safety Features Actuation System
SLCS	Standby Liquid Control System
TDAFW	Turbine Driven AFW
TER	Technical Evaluation Report
THERP	Technique for Human Error Rate Prediction
UFSAR	Updated Final 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 Big Rock Point (BRP) Individual Plant Examination (IPE) submittal and the responses to the Requests for Additional Information [IPE submittal, RAI Responses]. This technical evaluation report adopts the NRC review objectives, which include the following:

To assess if the IPE submittal meets the intent of Generic Letter 88-20, and

To determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335.

A Request of Additional Information (RAI), which resulted from a preliminary review of the IPE submittal, was prepared by BNL and discussed with the NRC. Based on this discussion, the NRC staff submitted an RAI to Consumers Power Company (CPC) on January 29, 1996. A subsequent telephone discussion between BNL and the NRC on February 20-22, 1996 revealed the need for additional clarification, which resulted in another RAI being sent to the CPC on March 1, 1996. CPC responded to both RAI packets in a document dated April 4, 1996 (RAI Responses). This TER is based on the original submittal and the responses to the RAIs.

1.2 Plant Characterization

The Big Rock Point (BRP) Nuclear Power Plant is a 75 MWe, 240 MWth General Electric boiling water reactor (BWR). This is an early BWR design (BWR-1), in many respects dissimilar to the other operating BWRs, having a much smaller power output than later BWRs. The reactor coolant system (RCS) consists of the reactor vessel, main feedwater system, main steam system, external motor pump driven recirculation loops, a steam drum, an isolation (or emergency) condenser and interconnected piping. There are no jet pumps in the BRP reactor pressure vessel and the downcomers are external to the vessel.

The reactor is housed in a large dry containment (unlike any other operating BWRs), which is a steel sphere 130 ft in diameter, and having almost 1 million cubic feet of free volume. There is no suppression pool, and thus, from a Level 2 standpoint, the plant can be compared to a typical PWR. The containment free volume to core thermal power ratio is substantially higher than at a typical PWR (almost 5 times that of Zion), as is the free volume to core mass ratio (almost 2.5 times that of Zion), while the containment ultimate failure pressure is a little over half that of a typical PWR (79 psig). The plant is operated by Consumers Power Company of Michigan (CPC), and started commercial operation in December 1962. There are no other operating units on site.

Design features at BRP that impact the core damage frequency (CDF) relative to other BWRs are as follows:

1) Large primary water inventory relative to core thermal power and decay heat levels. Over 35,000 lbm of water covers the core in the reactor and the steam drum, following a reactor trip. Therefore, it would take 2 hours to deplete inventory to the top of the core even if no decay heat removal systems were to function. This is a positive feature relative to other BWRs.

2) Emergency condenser (EC) for high pressure core cooling and makeup. This system is similar to the

isolation condenser found at some other older BWRs and is automatically initiated when the primary pressure reaches 1435 psig. It can also be initiated manually from the control room or the alternate shutdown building. The system enables passive cooling of the reactor, without reliance on ac power. DC power is needed for initial opening of the EC isolation valves, as well as for the valves admitting makeup water to the EC shell side. The makeup can be provided by the demineralized water system (if ac power is available), the firewater system or by the portable diesel driven pump (distinguished from the fixed diesel driven fire pump). The time scales for success of various steps in EC operation are relatively long: with no makeup supply for shell cooling available, the emergency condenser operation can prevent safety valve lifting (setpoint at 1535 psig) for a period of 6.5 hours (an additional 3 hours would pass before the core is uncovered). With makeup available, Big Rock Point has sufficient dc capacity to cope with station blackout conditions for over a week. Makeup valve actuation can be initiated remotely from the control room or the alternate shutdown building. The dc battery supplying power to the EC makeup valves is located in the alternate shutdown building.

The emergency condenser contains two tube bundles, each capable of removing 100% of the decay heat.

In case of ATWS, emergency condenser can remove full power for 30 min, thus giving the operators additional time to respond.

These are positive features relative to most other BWRs.

3) The main condenser, which, in conjunction with the feedwater system, the circulating water system and the turbine bypass system can be used for reactor makeup/decay heat removal. It has three sources of water inventory: hotwell inventory, the gravity feed from the condensate storage tank and a connection from the firewater system. The redundancy in condenser makeup is a positive feature, however, hotwell inventory would only last 3 minutes at full power, a negative feature from reactor inventory makeup standpoint, but a positive feature from containment failure standpoint.

4) The firewater system, consisting of one diesel driven and one electric pump, can be used as part of the ECCS. This system can be used for low pressure injection (i.e., core spray), for cooling of the post incident (i.e., recirculation) heat exchangers, or for "fill the ball" (used when recirculation is not available). It can also be used for emergency condenser secondary side makeup, or to provide inventory for the main condenser. This is a positive feature relative to other BWRs.

5) Lake Michigan can be used as the ultimate heat sink. The fire system takes suction from Lake Michigan. This is a positive feature.

6) A portable diesel driven pump is available for emergency condenser secondary side makeup if all other methods fail. This is a positive feature.

7) In case recirculation is unavailable for continued core cooling, "fill the ball" can be used. This involves continuing in the injection mode, thus filling the spherical containment with water. The post-incident (or recirculation) system is initiated when the water level reaches the 587 ft elevation inside the containment, in order to preserve containment integrity. The maximum permissible containment water level is 596 ft, based on the design pressure (about midplane). Depending on how many pieces of ECCS equipment are operating, this level will be reached in between 6.3 and 21 hours after the initiator. Should the post incident system fail, fill the ball is used, as calculations show that the containment can really withstand fill up to about 620-636 ft level (depending on internal air pressure),

at which point injection is terminated and passive cooling via natural circulation and air cooling of the containment steel shell takes over. However, although the operators are trained in this procedure, no credit is given for fill the ball operation in the IPE. If credited, this would be a positive feature.

8) Except for the diesel driven fire pump and the portable diesel driven pump, all pumps are motor driven (no turbine driven pumps). This is a positive feature, according to the IPE.

9) The emergency ac power consists of one 200 kW emergency diesel generator, and one 250 kW, standby diesel generator. The diesel capacity is sized just for the electric fire pump and/or one CRD pump and/or demineralized system for emergency condenser shell makeup, clearly below what other BWRs' capability and thus a negative feature. The diesels could also be used to power the instrumentation and control system, for operation of the alternate core spray valves or to charge the various batteries. The EDG automatically starts on detection of undervoltage on the 480V safety bus. The SDG is manually started should the EDG fail. There is only one emergency bus serviced by the diesels, a negative feature. Equipment is manually loaded onto the emergency bus on an as-needed basis. The core cooling function can be accomplished without ac power.

   DC power is supplied by two battery banks, i.e., the normal and the alternate shutdown battery. The alternate shutdown battery supplies most post-initiator loads of interest (including emergency condenser makeup valves and EC level switch, the reactor level and pressure transmitters, and the MSIV power), and is sized for days' of blackout conditions, a positive feature. The normal battery supplies most of the instrumentation, control and annunciator loads and opening of the core spray valves and is good for about 4 hours in a blackout. Dedicated batteries are provided for other systems (the diesel generators and the reactor depressurization system). Control room instrumentation has an alternate source of power should dc power be lost in conditions other than a blackout (125 V ac). The emergency power system requires no support function, a positive feature. The EDG is provided with, but does not need room ventilation, for at least 24 hours (confirmed by actual test). The SDG is mounted in a trailer, with ventilation provided by opening the trailer doors.

10) Either CRD pump capacity is greater than the decay heat levels, a positive feature. There are two CRD pumps, the standby pump starts automatically on reactor trip. However, CRD pumps cannot be used in conjunction with safety valve cycling or actuation of the reactor depressurization system due to high temperature in the CRD room, a negative feature.

11) The instrument air system has three air compressors, with one being sufficient for system success. Apparently domestic water can be used for backup cooling of air compressors, a positive feature. The instrument air system supports the main condenser (including hotwell makeup from the CST) and one method of emergency condenser shell side makeup (from the demineralized water system).

12) The RCW (reactor cooling water, similar to CCW at a PWR) system is not qualified for harsh in-containment conditions, a negative feature, and is thus not credited in case of LOCA/steam line break, or operation of safety valves or the reactor depressurization system. The system of interest which is cooled by the RCW is the shutdown cooling system. Firewater can be used as backup for RCW, however this is not credited in the IPE.

13) The SW system cools the RCW heat exchangers as well as the instrument air compressors and after coolers and the feedwater pumps (lube oil and seal coolers). Relatively few systems need service water.

14) Big Rock Point has a fast acting, manually actuated, passive liquid poison system or standby liquid system, utilizing nitrogen accumulators and squib valves. Reactor shutdown is achieved within 75 seconds of initiation (on first pass through the core). This is a positive feature.

15) The plant has a "100% load rejection capability", a positive feature. In the past, only load rejection from 50% power or lower has been successful. At higher power levels, load rejection has always failed due to secondary instabilities. However, prior to the IPE submittal, a hardware modification had been implemented, which seems to give the utility a high confidence (90% probability of success in the IPE) that load rejection from full power would be successful (no actual events had been experienced at the time of the submittal or the RAI responses). This modification is an automatic trip of one of the two recirculation pumps on load rejection. In case of load rejection, most of the steam would be bypassed to the condenser and the reactor power would be reduced (by increased voiding in the core, and later, by optional manual operator control rod insertion), such that the reactor and the turbine generator would continue to run, generating just enough electrical power for the house loads.

16) The 100% turbine bypass capacity, i.e., no safety valve challenge after failure to scram from full power under certain conditions, a positive feature. However, the feedwater system will not continue to operate unless a recirculation pump trip occurs (for the same reasons as in the paragraph above), lowering the reactor power by 40%. Thus, in practice, the plant really doesn't have the 100% bypass capacity (not for a long time at least). This feature is credited in high pressure sequences where the emergency condenser setpoint is reached and the recirculation pump is tripped automatically (as explained above). In low pressure sequences (e.g., spurious open condenser bypass valve), the operator has some time to manually trip one recirculation pump, however this is not credited.

17) No high flow rate high pressure ECCS pumps, a negative feature, except in case of ATWS where this may prevent containment failure.

18) A single two train low pressure ECCS for LOCA evolutions, a negative feature.

19) A portion of primary system piping which is located below core midplane, a negative feature.

20) No sign of the classic symptoms of IGSCC (inter granular stress corrosion cracking) found at other BWRs, a positive feature.

21) 200% full steam flow primary system safety relief capacity, a positive feature.

22) No total dependence of engineered safety features on support systems such as service water and instrument air, a positive feature. However, many systems are vulnerable to harsh environmental conditions, a negative feature (CRD, RCW, emergency condenser outlet valves, primary core spray valves, reactor level and pressure instrumentation, etc.).

23) The large dry containment which effectively decouples containment considerations from the Level 1 analysis, a positive feature.

24) The BRP sump which is less susceptible to strainer blockage than at other BWRs, a positive feature. This is due to: a lack of suppression pool (which may contain LOCA generated debris), as water is pumped from Lake Michigan; recirculation is not needed for a long time (and injection can be reentered should recirculation fail); there is a (non-proceduralized) capability to backflush the strainers; the post-LOCA pool is relatively large and stagnant, with relatively low flow velocities; the

suction strainers are located away from the containment floor and away from the top of the pool; and coarse mesh screen doors are located at the entrances of the containment areas with the suction strainers.

25) The RDS (reactor depressurization system) valves are dc power dependent. However, they have their own dedicated batteries (good for 5 hours in a blackout), a positive feature, independent of the normal station battery or the standby battery. The RDS is of limited help in a blackout due to the limited battery life and also because the core spray system depends on the normal station battery which has a life of only 4 hours in a blackout. This is a slightly negative feature (as the blackout is not a significant CDF contributor).

26) A relatively wide gap exists between the operating pressure of 1335 psi and the design pressure of 1700 psi, a positive feature.

27) In case of ATWS, and if stable conditions at full power have been achieved, an alternate, albeit time consuming method exists for shutting down the reactor (a positive feature). This is accomplished by batching the borax or boric acid into the condensate makeup system, which is then injected by feedwater.

The Big Rock Point (BRP) Plant utilizes a spherical steel vessel for a large dry containment. The plant is designed such that operating personnel may enter the sphere and remain inside as necessary during normal operation, shutdown, and refueling. Some of the plant characteristics important to the back-end analysis are summarized in Table 1 of this report.

Table 1 Plant and Containment Characteristics for Big Rock Point Plant

Characteristic	BRP	Zion	Surry
Thermal Power, MW(t)	240	3236	2441
RCS Water Volume, ft^3	600	12,700	9200
Containment Free volume, ft^3	940,000	2,860,000	1,800,000
Mass of Fuel, lbm	28,600	216,000	175,000
Mass of Zircalloy, lbm	11,480	44,500	36,200
Containment Design Pressure, psig	27	47	45
Median Containment Failure Pressure, psig	79	135	126
RCS Water Volume/Power, ft^3/MW(t)	2.5	3.9	3.8
Containment Volume/Power, ft^3/MW(t)	3917	884	737
Zr Mass/Containment Volume, lbm/ ft^3	0.012	0.016	0.020
Fuel Mass/Containment Volume, lbm/ ft^3	0.030	0.076	0.097

Because BRP is the only BWR plant that uses a large dry containment, it is more appropriate to compare the containment characteristics with those of PWRs with large dry containments. As seen in the above table, the thermal power level of BRP is more than 10 times smaller than those of Zion and Surry. On the other hand, the containment free volume of BRP is only about 2 to 3 times smaller. The containment volume to thermal power ratio, which is an indicator of the containment performance in meeting the pressure challenges during a severe accident, is much greater for BRP than for Zion or Surry. The data in the above table also show the

comparison of some other parameters, and all of these comparisons reflect the relatively large containment volume for BRP than for other plants. On the other hand, the containment pressure capability for BRP is lower. 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:

- The only BWR plant that uses a large dry containment (a spherical steel vessel).

- A containment that can be accessible during power operation. The plant is designed so that operating personnel may enter the sphere and remain inside as necessary during normal operation. The potential for a single access door to be open while personnel are entering or leaving the containment thus exists. Interlocks on the doors prevent simultaneous opening of both equipment lock doors or both personnel doors.

- The small core and large containment volume. The containment volume to core thermal power ratio is about 5 times that of PWRs with large dry containment. The large containment volume reduces the challenges to containment integrity from containment pressurization mechanisms. It also provides significant passive heat removal capability through the containment shell and other passive heat sinks.

- The capability to flood the containment. Procedures are in place at BRP to direct the operators to fill the containment vessel with water (called "fill-the-ball" in the IPE submittal). This provides cooling to the core debris in-vessel such that vessel failure may be avoided, or provides cooling and scrubbing of the debris ex-vessel if vessel failure is not prevented.

- A sump beneath the reactor vessel that has the volume to hold the entire core debris. The sump in the CRD room beneath the reactor vessel has a depth of 3 feet and a volume of 126 cubic feet. This sump may hold the entire BRP core after vessel breach (with a total depth of 1.6 feet, or 50 cm).

2 TECHNICAL REVIEW

2.1 Licensee's IPE Process

2.1.1 Completeness and Methodology

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

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

Internal initiating events and internal flooding were considered. Event trees were developed for all classes of initiating events. No uncertainty analysis was performed. Several sensitivity analyses were performed (with regard to the diesel fuel oil supply, the load rejection assumption, the hotwell makeup, the electrical bus failure rate, the temperature in Rooms 418 and 400, and the liquid poison squib valve sensitivity). System importance analysis was also performed (utilizing the Fussel-Vesely and the Birnbaum importance measures).

The IPE Level 1 model (submitted in late 1994) is an update of an earlier BRP Level 3 PRA, which was submitted in 1981 (for TMI exemptions) and reviewed by the NRC. Model updates reflect the plant modifications and data since 1981. The event trees were revised based on Rev. 4 of BWROG EPGs and updated success criteria. Other PRA studies were also reviewed: NUREG-1150 for Surry, Peach Bottom and Grand Gulf, and the 1983 Shoreham PRA. It seems that the Sequoyah PRA was also reviewed.

The submittal information on the HRA process was adequate. However, on the basis of the licensee's response to the NRC's RAI, information in some parts of Section 4 (the overall "methods and approach") was apparently inaccurate. The response to the RAI indicates that some of the information in section 4 (at least some related to the HRA) was *boilerplate* and should have been revised after completion of the analysis. Sections 10 and 13 provided enough information to evaluate the HRA, but it would have helped to be able to rely on the information in section 4. Nevertheless, the information contained in the submittal and that obtained from the licensee's response to the RAI, indicated that the HRA was generally complete in scope.

The HRA process for the Big Rock Point 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). A detailed analysis was performed for all pre-initiators. The Big Rock Point IPE acknowledges both response and recovery type post-initiator human actions. However, post-initiator actions were modeled only when clear procedural guidance (normal, abnormal, or emergency procedures) existed for the operators and repair activities were apparently not credited. To account for dependencies during the initial (screening) analysis, the submittal states that "where it was initially recognized that resulting sequences may contain multiple operator actions, the HEPs were initially set to 1.0." After the initial quantification and when quantified operator actions were first included, the nominal ASEP HRA method was applied to all post-initiator human actions. Where important sequences contained multiple operator actions, the actions were analyzed to determine the dependencies between the HEPs. The HEPs obtained using the ASEP method are known to be somewhat conservative. After the sequences were quantified with the ASEP values, operator actions identified as potentially being important were re-analyzed using the THERP methodology. All the actions re-analyzed had a Birnbaum importance greater than 1.0E-6.

While in many cases the application of THERP was reasonable, there were several events for which the quantification process did not seem appropriate. It is thought that the resulting HEPs should be considered optimistic and that the use of such values for these events is a weakness of the HRA. The problem arises through the licensee's use of HEP values from the "annunciator response model" (Table 20-13 or Table 11-13 from THERP) in situations where very limited time is available for the operator action. While it can be argued that the HEPs from this model are acceptable when substantial time is available for the operators to determine the relevant actions and when the operators need only respond to the existence of an annunciator in the control room, the HEPs from this model do not reflect the impact of the time available on the likelihood of success. Thus, this model will clearly underestimate HEPs for short time frame scenarios and the ASEP/THERP time-reliability diagnosis model is clearly indicated in such situations. Other important aspects of the post-initiator HRA analysis appeared to be conducted appropriately. Human errors were identified as important contributors in accident sequences leading to core damage and several procedure related enhancements were discussed in the licensee's response to the NRC's RAI.

The Big Rock Point Plant Individual Plant Examination (IPE) back-end submittal is essentially consistent with respect to the level of detail requested in NUREG-1335.

The methodology employed in the BRP IPE for the Level 2 evaluation is clearly described in the submittal. Plant Damage States (PDSs), which are defined in the IPE by an event tree structure with the parameters important to Level 2 accident progression as the top events, are defined in the IPE. They are further grouped to key PDSs (KPDSs) to be used as the initial conditions for the Level 2 analysis. Quantification of the Level 2 accident progression involves the development of small top level containment event trees (CETs). The top events of the CETs are determined by the fault trees, the PDS definition, and analyses of important containment phenomena. The CETs and the supporting logic trees addressed in detail all the containment failure modes discussed in NUREG-1335. The results of the CET analyses are an extensive number of CET end states which are binned into fifteen release categories (only five of which have non-zero frequencies). The CET quantification relies on review of industry literature, primarily the NUREG-1150 document, and plant-specific analyses using the MAAP-BRP code. Release fractions for the release categories are based on the plant-specific MAAP-BRP calculation results. However, only release fractions for CsI are reported in the IPE submittal, The release fractions of other fission product groups are not reported in the submittal.

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

There are no other units on site.

A wide variety of up-to-date information sources were used to develop the IPE: Final Hazards Analysis Report (for system success criteria), BRP technical specifications (system operating guidelines and system design), plant operations manual (system descriptions and operating procedures), emergency operating procedures (system operation during an emergency and operator actions during an emergency), BRP drawings (system layout, system interconnections and component control schemes), scram reports, event reports and licensee event reports (initiating event data, plant response and failure data), plant surveillance procedures (demand data, test frequencies and run times) and maintenance orders (failure data and component availability).

The plant configuration was modeled as it existed early in 1993. The data was collected for the last 10 years of operation, while the initiating event frequency data were collected from 30 years of operating experience. Many plant walkdowns have been performed throughout the 30 year BRP history. In addition to the walkdowns performed for the 1981 PRA, the various plant upgrades and other plant specific activities, several

walkdowns were performed as part of the IPE. These were performed on an as needed basis as part of the fault tree and event tree development, and also for the flooding, HEP and containment analyses. The walkdowns were part of the iterative PRA process, and were easier to perform here than at most plants because a) the PRA team (i.e., CPC personnel) work on site and b) the BRP containment is not inerted and access is relatively easy.

Significant participation in the IPE by plant staff, procedure reviews, discussions with operations and training staff, observations of simulator exercises, review of the "control room design review", and walkdowns of important operator actions, including local actions, helped assure that the IPE HRA represented the as-built, as-operated plant. Contractors (not named) and training, operations and management personnel performed review of the HRA. This also helped assure that the IPE HRA represented the as-built, as-operated plant

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

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

2.1.3 Licensee Participation and Peer Review

Licensee personnel were involved in all aspects of the analysis. In-plant expertise was already existent due to the previous BRP PRA study, such that CPC personnel performed most of the work, with (unspecified) help from Gabor Kenton and Associates, and Tenera. It appears that almost all of Level 1 work was done by CPC.

The reviews performed for the IPE seem to have been done by the analysts in the course of their work and by other CPC personnel in the course of interactions with the maintenance, engineering, reactor engineering and training personnel. Formal review was performed by the present plant manager, operations manager, the simulator operations supervisor, the safety and licensing director, selected SRO qualified training nuclear instructors and selected maintenance personnel (RAI responses).

Outside review was performed by experts from Tenera, Gabor Kenton & Assoc. (now Dames & Moore) and by an independent contractor (David Bizzak).

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 initiating events for BRP IPE were identified primarily based on the 1981 PRA and by reviewing the plant operating experience. A couple of minor initiators from the 1981 PRA were deleted as they would not be a significant contributor. Controlled manual shutdowns were not included in the analysis, but forced manual shutdowns were, according to the submittal, although the value of 5.6 forced manual shutdowns per year appears high.

As a result, a total of 35-40 initiating events were identified (some initiators, such as ISLOCA or turbine trip can be further broken down into subinitiators). In addition, only 1 flooding scenario survived the screening process, and will be described in the flooding section of this report. The internal initiators are:

LOCAs:
- RPV Rupture below core
- Large LOCA below core
- Medium LOCA below core
- Small LOCA below core
- Very small LOCA below core
- Large LOCA above core
- Medium LOCA above core
- Small LOCA above core

Steamline break inside containment:
- Large SLBIC
- Medium SLBIC
- Small SLBIC
- Very small SLBIC

Steamline break outside containment:
- Large SLBOC
- Medium SLBOC
- Small SLBOC

Transients:
- Turbine Trip
- Loss of Feedwater
- Loss of Main Condenser
- Spurious MSIV Closure
- Spurious Bypass Valve Opening
- Spurious RDS (Reactor Depressurization System) Valve Opening
- Load Rejection
- Loss of DC Power
- Loss of Instrument Air

ATWS:
- Turbine Trip
- Loss of Feedwater
- Loss of Main Condenser
- Spurious MSIV Closure
- Spurious Bypass Valve Opening
- Loss of Instrument Air
- Loss of Offsite Power

Special Initiators:
- Interfacing Systems LOCA
- Internal Flooding

Manual Shutdown:

Manual Shutdown
Loss of Service Water

The initiating event list seems to be complete and comparable to events considered in other PRAs. HVAC failures do not lead to initiating events (RAI responses).

The loss of offsite power initiating event is included under "load rejection". Load rejection occurs when the main 138 kV transmission line "disconnects". At that point, the plant attempts to continue running the reactor to just supply the house loads (i.e., one recirculation pump is tripped off, voiding in the core increases, most of the steam is bypassed to the condenser, with only a small fraction flowing to the turbine in order to supply the 4-6 MWe needed). If this is unsuccessful, the turbine will trip and the reactor will scram (e.g. on high flux), and transfer of essential loads to the 46 kV transmission line will be attempted, automatically. (Large loads, such as the feedwater pumps and the reactor recirculation pumps would be tripped automatically, with manual loading possible if the transfer to the 46 kV line was successful). Only if this transfer is unsuccessful, will the "loss of offsite power" occur, and the emergency diesel generator will start to supply select safety loads (which would be loaded manually), mainly the electrical fire pump and/or the CRD pumps.

The loss of DC power initiator involves loss of power from the normal station battery (there is only one normal DC bus in this plant). The standby battery powers certain safety loads (e.g., the emergency condenser makeup and isolation valves), and its failure would not constitute an initiator (although LCO conditions would be entered, and the plant would have to be shutdown in 24 hours).

Loss of an individual AC bus was considered, and frequency of individual faults calculated. Apparently, this was not an important initiator (highest fault frequency was 7.7E-5/yr), and thus is not included in the Table of initiators, (Table 4 in Section 2.2.2.6).

For a discussion of initiator frequencies see Section 2.2.2.6 below.

2.2.1.2 Event Trees

The IPE developed 37 event trees to model the plant responses to internal initiating events. Pretty much every initiating event has a separate event tree developed for it. In case of interfacing systems LOCA, there are two event trees, one for each of the two dominant pathways, and in case of load rejection there are three, depending on the stage in progression from failed load rejection to station blackout. Thus there are 7 general transient event trees (turbine trip, manual shutdown, loss of feedwater, MSIV closure, loss of main condenser, loss of instrument air and spurious bypass valve opening event tree); 8 steam line break event trees (the 4 categories inside the containment, the 3 categories outside the containment plus the spurious RDS operation event tree); 7 LOCA event trees (corresponding to the three categories above the core and 4 categories below the core, no event tree is developed for the RPV rupture; however, all below LOCA events are treated as RPV ruptures at the lowest point in the primary system, including the large LOCA, with a break area of up to 3.5 ft^2; 4 loss of power event trees (failed load rejection, loss of station power, station blackout and loss of DC power event trees); 7 ATWS event trees, corresponding to the 7 ATWS categories enumerated in the initiator section; and 4 "other transients" event trees (the internal flooding event tree, the two interfacing LOCA event trees and the long term cooling event tree). The long term cooling event tree is a transfer event tree used for transients where low pressure conditions for initiation of the shutdown cooling system or the post incident system exist within the 24 hour mission time.

The event trees are systemic. The mission time used in the core damage analysis was 24 hours.

The event tree end states are divided into two possible outcomes: success or core damage (which is then put into the appropriate plant damage bin).

The analysts used the peak clad temperature of 2500°F as the definition of core damage.

Success criteria are based on performance or review of engineering analyses. These analyses were comprised of the following:

1) Existing in-house best estimate or design basis calculations performed for Big Rock Point;

2) Hand calculations tailored to the event sequence success criteria development, based on continuity, momentum and energy balances;

3) Computer modeling of the event sequence success criteria using plant analysis codes such as MAAP;

4) Engineering judgement.

The success criteria appear reasonable. The licensee claims that the RDS success criterion of 3 out of 4 valves opening for depressurization, used in the IPE, is conservative. In reality, in all but two sequences, one out of 4 valves is sufficient (RAI responses). In all non-ATWS scenarios, lifting of one out of 6 safety valves (different system from the RDS) will prevent overpressurization; in ATWS that success criterion is 3 out of 6 safety valves.

The RCP (recirculation pump) seal LOCAs are not modeled, either as an initiator or as part of an accident sequence, for the following reasons:

1) Procedures instruct the operators to trip the RCPs if seal pressure and temperature cannot be maintained;

2) RCP loop isolation valves could be closed to isolate excessive seal leakage if ac power is available;

3) If seal cooling were lost via loss of reactor cooling water or loss of service water, RCP seal failure would not occur immediately. A test of the BRP RCP O-rings conducted in 1981 exposed the seals to 580°F for a 6-hour period without failure.

The steam drum enclosure sprays are modeled for certain initiators (e.g., steamline breaks inside containment) because of environmental operability requirements of certain equipment within the enclosure (e.g., emergency condenser valves).

The environmental qualification requirements of systems are noted and modeled (e.g., harsh LOCA conditions inside the containment; flooding of containment by injection systems or from steam condensation, etc.). However, the effect on terminal connections and cables is apparently not modeled.

No repair activities were credited except for offsite power recovery. Certain recovery actions (e.g., condenser hotwell makeup via firewater in case of ATWS) were not credited. The fill-the-ball operation does appear in the event trees but is assigned a failure probability of 1. There is also a discussion which considers that cooling of the core through nucleate boiling via a submerged lower head of the reactor vessel should be effective, but this is not credited, either. On balance, these conservatisms seem reasonable (based on limited time available, lack of thermal hydraulic discussion etc.). The assumptions related to the fill-the-ball strategy

always failing are responsible for 42% of the core damage. This is due to LOCA sequences. Thus, if this strategy were credited, the CDF would drop by almost a half, but the LOCAs would still be a dominant contributor, therefore the conclusions regarding relative importance of various accident scenarios would still be mostly valid.

Other modeling conservatism are also no d in the RAI responses. Feedwater hydro of the steam drum was not credited (the operators could raise the steam drum level to above the steam separators and cool the core through the risers for the small pipe break initiators). Emergency condenser could be used for depressurization in very small LOCA below core conditions below the shutoff head of the fire pumps (not credited). The proceduralized manual operation of the alternate core spray injection path (MOV-7072) was never modeled. The proceduralized portable diesel pump was only credited in case of station blackout and internal flooding. The recently installed fourth air compressor was not credited.

For ISLOCAs pathways outside containment through low pressure piping were considered. Inside containment ISLOCAs were included in the LOCA and steam line break inside containment analyses. Outside containment steam line breaks were also considered in a separate analysis (see above), therefore ISLOCA analysis does not consider such events (including, for example breaks in emergency condenser tubing).

Two dominant pathways resulted from this analysis, and two separate event trees were developed. One was an ISLOCA above core, which resulted in the core spray injection line. The structure of this tree is similar to that of a medium LOCA above core, with some events missing: the core spray is assumed failed, the enclosure spray is not needed, the post incident and the fill-the-ball systems are not credited as the discharge is outside the containment. The other event tree was for an ISLOCA below the core, which represented a break in the reactor fuel pit drain/blowdown line. The structure of the tree is similar to that of a small LOCA below the core without events representing the post incident system and the fill-the-ball strategy.

2.2.1.3 Systems Analysis

A total of 22 systems/functions are described in the Submittal. Included are descriptions of the following systems: emergency condenser, emergency condenser makeup, reactor depressurization system, primary system safety relief valves, main steam isolation system, main condenser, shutdown cooling system, post incident system, feedwater system, condensate system, control rod drive system, fire protection system, core spray system, enclosure spray system, containment isolation system, fill-the-ball, station power system, component and instrument air, condenser circulating water system, reactor cooling water system, service water system and liquid poison system. A discussion of the HVAC system considerations was provided in the RAI responses.

Note: the post incident system is analogous to a low pressure recirculation system in a PWR, i.e., suction is taken from the containment sump via 5 strainers, through two parallel 100% capacity pumps outside the containment, through a heat exchanger and then back inside the containment and over the core through spray spargers.

Each system description includes a discussion of the system description and function, support systems, testing and maintenance, technical specifications and requirements, system operation under accident conditions, system success criteria, key modeling assumptions, fault tree description, common cause information (including the values of the MGL parameters), human reliability, plant specific experience, results and insights (including the Fussell-Vesely and the Birnbaum importance measures for the system) and initiating event review.

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

Section 1.2 of this TER provides a description of important plant features.

2.2.1.4 System Dependencies

The IPE addressed and considered the following types of dependencies: shared component, instrumentation and control, isolation, motive power, direct equipment cooling, areas requiring HVAC, operator actions and environmental effects. The HVAC system is assumed to be not needed in any of the rooms, due to tests showing equipment not reaching the damage temperature (e.g., in the control room), availability of natural circulation and passive cooling due to non-compartmentalized nature of certain areas, and availability of simple operator actions such as opening of a roll up door.

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. The event trees were systemic. The IRRAS computer code was used for development and quantification of top event probabilities and accident frequencies.

The cut set truncation limit used was 1.E-09/yr, without the initiating event frequency. Initial cutsets with a conditional probability of 1.E-6 were scrutinized to make sure they made sense.

The IPE took credit for recovery of offsite power. The IPE power recovery curve is based on BRP experience and is consistent with average industry data cited in an Electric Power Research Institute (EPRI)-sponsored study (NSAC-147) (actually the BRP data are conservative compared to the NSAC data).

No other recoveries or equipment repair were credited. For instance, restoration of the main condenser after MSIV closure was not credited, nor was the restoration of feedwater in a loss of feedwater event tree. In addition, no recovery was included in the Level 2 analysis. However, simple recoveries from operator errors were credited.

Other conservatisms in the model are related to double counting of data and cutsets. Minimal sorting of raw failure data was done, such that a failure of a supercomponent such as the diesel generator may also be counted as a failure of its constituent parts, e.g., a relay. Non-minimal cutsets were not eliminated, as an older version of IRRAS was used (Ver. 4.16) which apparently did not have that capability, according to the submittal[1].

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. The CDF

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[1] This last assertion, about IRRAS version 4.16, may not be correct. The author of this TER tried to contact Ken Russell of INEL, the developer of IRRAS, unsuccessfully, but the author's recollection is this capability existed much sooner than this version, and, definitely, much earlier than 1993, the year the IPE was done (the earlier versions of IRRAS, i.e. 2.xx, back in 1989 and 1990 had this problem).

calculated is 5.4E-5/yr. No uncertainty analysis was performed. Fussel-Vesely and Birnbaum importance measures are given for systems and key basic events, and insights are gained about importance of systems and maintaining their performance based on these measures.

Six non-HRA sensitivity studies were also performed.

In the first study, sensitivity to the diesel fuel oil supply is explored. There are several diesel engines on site: the diesel driven fire pump, the emergency diesel generator, the standby diesel generator and the portable diesel pump. All of these motors, except for the portable pump, require dedicated fuel. It is possible to refill all the tanks with the same fuel, thus incapacitating the engines for which this is the wrong type of fuel. The sensitivity study set all the diesel engine run failure probabilities to 1. The increase in the CDF was 2.25E-3/yr, i.e., very substantial. This is understandable, as these diesel engines are used as backup in many scenarios.

The second study calculated the sensitivity to the load rejection assumption. The current assumption is that the load rejection from full power would fail only 10% of the time, due to the recent hardware modification of tripping one recirculation pump on load rejection. However, this assumption is untested (there have been no load rejections since the modification), and full power load rejections had always failed prior to this improvement (partial power load rejections, from up to 50% power had been successful in the past). This study set the load rejection failure probability equal to 1. The resultant increase in the core damage frequency was 6.0E-6/yr, i.e., only about 11%. This is due to the electrical distribution system redundancy, existence of two diesel generators, the diesel driven fire pump and the portable diesel pump, as well as the long time to battery depletion (about a week). It seems plausible that this sensitivity to the load rejection assumption would be increased with increased unreliability of the two diesel driven pumps (a point of one of the RAIs, see section 2.2.2.3), however no updated sensitivity study was performed as part of the RAI responses.

The third sensitivity study was performed for the ability to use the condensate system as a low pressure water supply to the reactor. Makeup to the condenser hotwell can be provided by gravity feed from the condensate storage tank or by the fire protection system (the latter was apparently not credited due to the short time scale for this manual action). The condenser makeup was disabled in this study (i.e., failure probability of makeup valves set to 1), thus disabling the low pressure condensate pump injection into the reactor, the resultant increase in the CDF was 7.6E-5/yr, i.e., significant. In this case, the low pressure makeup to the reactor vessel is reduced to the design basis (the core sprays with fire water). When the valve failure rate was increased by an order of magnitude from its base case value, the increase in the CDF was 6.0E-6/yr, i.e., relatively small (11%). Thus the reliability of these components can be relatively flexible.

The fourth study calculated the importance measures (Fussel-Vesely and Birnbaum) of various electrical distribution panels and buses. The Fussel-Vesely importance for all of these components was relatively low (between 1.E-5 and 6.E-3). The Birnbaum importance ranged from medium (1.4E-5/yr) to high (7.5E-3/yr), i.e., these would be the increases in the CDF if the appropriate component were assumed failed. The highest Birnbaum importance was derived for the panel that would service the post incident system, the manual pressure control and the condensate valve control.

The fifth study dealt with the temperature in Rooms 418 (spent fuel pit heat exchanger room) and 400 (steam drum enclosure). The LOCA and steamline break event trees were reanalyzed assuming increased failure probabilities of equipment within these two rooms due to severe environments. The important equipment in these rooms are the core spray level and pressure instrumentation used for automatic valve operation (room 418), reactor water level transmitters for the low reactor water level permissive signal for the RDS (room 418), the primary core spray motor operated valves (room 400) and the emergency condenser outlet valves.

Two sensitivities were performed, one assuming a moderate degradation of equipment performance that doubles the base failure rate, and two, a severe degradation which increases the base failure rate by an order of magnitude. The moderate degradation case produced a CDF increase of 2.9E-6/yr, which is small, i.e., moderate degradation can be tolerated. The severe degradation increased the CDF by 3.8E-5/yr, which is significant.

The final study considered the sensitivity of the liquid poison squib valves to thermal degradation while in service (these valves are regularly replaced every few years, and randomly tested). The seven squib valves are explosive actuated valves which have replaceable primer and trigger assemblies. Two sensitivity studies were performed. The first study assumes the squib valves would fail to actuate when exposed to a high temperature steam environment, as would be expected in case of safety valve actuation during an ATWS event. The second sensitivity assumes the squib valves will automatically actuate when exposed to prolonged (greater than 10 minutes) high temperatures. The first sensitivity produced a small increase in the CDF of 2.0E-6/yr. This is due to the fact that majority of ATWS sequences arise from the low pressure turbine trip, thus no SRV actuation results (ATWS contributes 7% to the CDF overall). The second sensitivity produced a small decrease in the CDF of 6.0E-7/yr, as the operator action to fire the squib valves is effectively removed.

2.2.2.3 Use of Plant Specific Data

The data collection process period was from 1982 through 1992 (10 years). The initiating event collection period was from the start of commercial operation in 1964 through the end of 1992 (29 years).

Both demand and time related failures were addressed. The sources of plant specific failure data and maintenance/testing unavailability data were: control room log books, deviation reports, licensee event reports, maintenance orders, surveillance tests and switching and tagging orders.

For components for which no plant specific data existed, generic data were used. For components for which relatively plentiful plant data existed, that data was used directly (e.g., dividing the number of failures by the 10 year data window and adjusting for the plant capacity factor). For components for which relatively sparse data existed, Bayesian updating of generic data with the plant specific experience was performed, according to the submittal. however, a spot check of the data base leaves the impression that Bayesian updating was used extensively (e.g., in case of the fire pumps, relatively plentiful plant specific data exists, yet Bayesian updating was performed). Most important components have plant specific data.

The submittal shows both the generic data and plant specific data used for a component, along with the plant specific experience (e.g., number of failures and total running time in hours) for that component.

Table 2 of this review compares the plant specific failure data for selected components from the IPE to values typically used in PRA and IPE studies, using the NUREG/CR-4550 data for comparison [NUREG/CR 4550, Methodology].

BRP data are generally in agreement with the NUREG/CR-4550 data, with data for the standby diesel generators and the diesel driven fire pump lower than expected. The reported failure rates seem to be supported by the exhibited plant data and the generic priors used, except for the fire pumps, see discussion below and in Section 2.2.2.4. Note that in Table 2, some failure rate data under BRP the column are generic, i.e., there was insufficient plant specific experience (e.g., electric bus, squib valves).

Table 2 Comparison of Failure Data

Component	BRP	4550
MD Pump fail to start fail to run	 4.0E-4 to 8.0E-3 1.7E-5 to 2.5E-4	 3.0E-3 3.0E-5
Electrical fire pump fail to start fail to run	 1.2E-3 3.6E-4	 3.0E-3 3.0E-5
Diesel Driven fire pump fail to start fail to run	 2.4E-3 4.8E-4	 3.0E-2 8.0E-4
Diesel Driven portable pump fail to start fail to run standby failure (90 days btw. tests)	 7.0E-2 (HEP) 3.4E-5 3.4E-5	 - 3.0E-2 8.0E-4
IAS Compressor fail to start fail to run	 4.3E-3 9.5E-5 to 1.4E-4	 8.0E-2 2.0E-4
Battery Charger Failure	1.9E-5 to 5.2E-5 (1.3E-5 for RDS chargers)	1.0E-6
Battery Failure	2.0E-6	1.0E-6
Circuit Breaker (480V) fail to remain closed fail to close	 2.2E-6 to 2.8E-6 3.6E-3 to 3.0E-2	 1.0E-6 3.0E-3
AC Bus Fault during operation	5.0E-7	1.0E-7
Check Valve fail to open fail to close	 5.3E-4 to 1.8E-3 6.1E-4 to 6.3E-4	 1.0E-4 1.0E-3
MOV fail to open fail to close	 2.7E-3 to 2.8E-2 1.2E-2 to 2.0E-2	 3.0E-3 3.0E-3
Air Operated Valve fail to open/close	 1.6E-3 to 2.6E-3	 2.0E-3
Solenoid Valve fail to operate	 1.3E-3 to 7.2E-3	 2.0E-3

Table 2 Comparison of Failure Data

Component	BRP	4550
Pneumatic valve, hydraulic valve fail to operate	1.5E-3 to 2.4E-3	2.0E-3
Explosive valve (squib)	3.0E-3	3.0E-3
Steam Drum Safety Relief Valves fails to open	4.6E-3	1.0E-5
RDS valve fails to energize/open	7.0E-3 to 1.5E-2 (solenoid + relief)	1.0E-2
Emergency Diesel Generator, Standby D.G. fails to start fails to run	1.6E-2, 4.4E-3 4.8E-2, 2.2E-2	3.0E-2 2.0E-3

(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.

In case of the diesel driven fire pump (DFP), there were two aspects of the data which were problematic. First, the generic prior used for Bayesian updating was the failure rates for the motor driven pump, whereas the diesel driven pumps tend to be significantly more unreliable. This would bias the posterior failure rate toward lower values, if the plant specific experience were weak (e.g., no failure). Second, while the plant specific experience was relatively strong, this was not reflected in the posterior values for DFP failure rates, which were unreasonably low.

Specifically, the generic value used for failure to start and failure to run, respectively, was 3.3E-3/d and 3.4E-5/hr, whereas the corresponding values in NUREG/CR-4550 are 3.0E-2/d and 8.0E-4/hr. The plant specific evidence for the DFP was 2 start failures in 814 demands and 7 failures in 407 hours of operation (i.e., testing). This yields the plant specific start failure rate of 2.5E-3/d and the run failure rate of 1.7E-2/hr. As can be seen from Table 2, the DFP run failure rate used in the IPE is much lower than expected (4.8E-4/hr), in spite of the relatively strong plant experience. Somewhat similar, though less pronounced, problem appears in the electric (motor driven) fire pump (MFP) failure to run (3.6E-4/hr used in the IPE vs. 4.4E-3/hr calculated from 3 failures in 679 hours). As far as the diesel driven portable pump is concerned, the usage of a low generic pump failure rate is somewhat offset by including the standby failure rate in the model.

The data used for these pumps is important as they are used for makeup to the emergency condenser, for the core spray injection and recirculation cooling, and would be depended on in a station blackout. As a result, a relatively high Birnbaum importance is calculated for these pumps (4.E-4/yr for the DFP and 6.E-4/yr for the MFP).

In response to the RAI dealing with these issues, the licensee stated that the plant data collected over predict failure for several reasons: the actual run time of the pumps during tests is sometimes much longer than the ½ hour required; the failures may be double counted between the pumps and their modeled constituent components (relays, switches, etc.); failure is assumed even if a slight departure from nominal parameters occurs, even if the pump can still produce the required flow (for example of the 7 DFP run failures, only one was claimed to be a "true" failure).

The licensee also states that other conservatism are built into the analysis which would offset any optimistic failure rates. The assumed mission time of 24 hours is reasonable for breaks below the core, but is conservative for most other initiators. For above core breaks and emergency condenser operation, the fire pumps would be operated intermittently. Also not included are passive containment sinks and core cooling via liquid film vessel cooling, which the CPC analysis shows would be effective in long term decay heat removal.

The licensee performed a sensitivity analysis in which the fire pump failure rates were raised to 1.0E-2/d for start failures and 1.0E-2/hr for run failures. The new total CDF is not shown, although it appears it is at least double the base case CDF (i.e., on the order of 1.E-4/yr). Instead, a comparison is made between CDF contributors (in terms of initiators) between the submittal and the sensitivity analysis, in which three types of initiators are compared: LOCAs and SLBs, loss of instrument air, and other non-LOCA. In the sensitivity analysis, the LOCA and SLB contribution as a fraction of the total CDF dropped precipitously, from 80% in the base case to 46%. The loss of instrument air contribution rose markedly, from 9% in the submittal to 42% in the sensitivity analysis. The other non-LOCA contribution stayed approximately the same, in a relative sense (11% of the CDF in the submittal vs. 12% in the sensitivity analysis).

There are four sequences in the loss of instrument air event tree which cause most of the increase in this initiator contribution to the total CDF. All four have failure of the emergency condenser and low pressure makeup to the core, which are dominated by failure of the fire pumps to run. The licensee states that, in reality, the recently installed fourth air compressor would also be used, and also the portable diesel pump would be used for emergency condenser makeup. The portable pump is now credited only for station blackout and flooding sequences. The fourth air compressor was not credited, however it has not been formally related to plant operations (RAI responses). According to the RAI responses, including the portable pump in the analysis would reduce the loss of instrument air contribution to 7% of the new CDF, while including the fourth compressor would reduce this even further. It should be noted that the portable pump run failure rate used is also optimistic, by 1 to 2 orders of magnitude. However, due to inclusion of the operator start failure rate of 7.E-2/d and the standby failure rate which adds another 3.7E-2/d to the start failure rate, this pump's mission time total unavailability is only mildly under represented.

2.2.2.4 Use of Generic Data

Generic data used were mostly PLG data, with some EG&G data and data from the Palisades and the Monticello IPEs/PRAs. The generic data are generally reasonable, with the exception of data used for the diesel driven pumps. In that case, the generic "pump" is used, i.e., data for the motor driven pump are used. This will lead to underestimation of failure probability of the diesel driven fire pump (generic data used as prior for Bayesian updating). In turn, this may lead to under representation of station blackout sequences in the list of dominant sequences. The licensee did perform some sensitivity studies to correct this deficiency (see discussion in section 2.2.2.3 above), in response to the RAIs.

2.2.2.5 Common-Cause Quantification

Redundant components were systematically examined to address potential common-cause failures. The approach used was the multiple Greek letter approach (MGL). The β and (if applicable) the γ, the δ, the ϵ and the ζ factors are reported in the submittal, with discrimination based on failure modes (e.g., in general, different values of MGL parameters are given to failure to start as opposed to failure to run).

The methodology used was not described in detail, although it appears that it has yielded reasonable common cause parameter values (for the most part). The process used is consistent with that described in NUREG 1478, "Procedures for Treating Common Cause Failures in Safety Reliability Studies" (RAI responses). The submittal states that the following references were consulted for data on common cause failures: EPRI NP-3967, the PLG data base, NUREG/CR-3289 and NPRDS.

When calculating the common cause failure rate of two or more components, the highest random failure rate is used as the basis for calculation. For instance, in case of the common cause failures of the emergency diesel generator and the standby diesel generator, the failure of the emergency diesel generator is multiplied by the β factor to arrive at the common cause failure rate. The EDG failure rates are higher (see Table 2).

As can be seen from Table 3, most important components are modeled with regard to the common cause failure. The batteries are not modeled, e.g., common cause between the station battery and the alternate shutdown panel (or standby station) battery, because they are located in different plant areas and are tested and maintained by separate procedures (RAI responses). Still, the batteries are of the same design, and some kind of common cause factor should have probably been included. (These two batteries are used for different purposes, there is only one safety dc bus, served by the alternate shutdown panel battery). The common cause among the four dedicated RDS batteries is not modeled, due to probabilistic insignificance, as the chargers can supply the needed power. There are other batteries in the plant, such as the ones associated with the diesel generators and the diesel driven fire pump, but apparently their failure is included under the supercomponent failure to start.

The common cause failures between the diesel driven and the electric fire pump are not modeled, even though the pumps are identical four-stage vertical Worthington turbines. Common cause failures due to environmental factors (e.g., flooding), maintenance errors, functional similarities (e.g., plugging by the zebra mussell) are included in the fault trees. Common cause due to design similarities would manifest itself in the metallurgical defects of the driver, which have not manifested themselves during routine disassembly of the pumps. However, lack of observed defects or observed failure mode does not necessarily mean lack of common cause failure between components. Common cause between pumps using different drivers has been modeled in other PRAs (e.g., between the motor driven and the turbine driven AFW pumps in PWRs). Due to the significance of the fire protection system as a frontline ECCS system, it appears there is a slight undercounting of failure in this regard. However, there seems to have been communication between the independent reviewer and the analyst on this issue (at first, CCF of these two pumps was modeled).

A comparison of effective β factors in the submittal vs. those suggested in NUREG/CR-4550 ("reference β factor") is presented in Table 3. Note that NUREG/CR-4550 does not distinguish between failure modes (e.g., failure to run vs. failure to start) in common cause failures, and the values given are those for failure to start.

The table shows general consistency between the BRP CCF data and that recommended in NUREG/CR-4550. For some pumps, the CCF data are somewhat lower in the IPE. The safety relief valve CCF values are substantially lower in the IPE. This is due to the fact that the BRP valves are of a different design; they are

spring loaded mechanical valves, whereas the NUREG-1150 BWRs use pilot operated relief valves (RAI responses). It is not explained how this affects the possibility of common cause failure between valves (as opposed to their failure rate). However, the impact of even including the higher CCF values is negligible on the final result (RAI responses). This is substantiated by the low Birnbaum importance calculated in the IPE for these valves (4.5E-7/yr for the whole system of 6 valves).

Table 3. Comparison of Common-Cause Failure Factors

Component	BRP β	4550 β
EDG, SDG (FTS&FTR)	0.05	0.038
pumps FTS FTR	 0.01 0.001	 0.056 to 0.21
SW pumps, FTS&FTR	0.03	0.026
CRD pump FTS FTR	 0.07 0.001	 0.21
AFW aux. oil pumps; core spray pumps; shutdown cooling pumps;FTS&FTR	0.11	0.11 to 0.15
Instrument air compressors FTS FTR	 0.07 0.01	
circuit breaker, FTC	0.07	
AOV	0.07	0.10
RDS relief valves	0.07	0.07 (?) (PWR PORVs)
solenoid vlv	0.07	
ck valve	0.01	
MOV, FTO&FTC	0.08	0.088
safety relief valves, FTO	0.01	0.22
transmitters (level, etc)	0.17 to 0.20	
switches	0.07	

2.2.2.6 Initiating Event Frequency Quantification

The initiating event frequencies used in the IPE are presented in Table 4, and the method used in calculation of the frequency is noted there as well. Following are some comments on the IE calculations and frequencies found in the Table.

For spurious RDS actuation, the potential for RDS opening due to valve failures and/or misalignment is calculated. It includes one event in 1993 (outside the data window) where one valve was found in the wrong position after replacement of a leaking pilot valve. The event did not result in an RDS valve opening, however it was counted as part of the cutset for this occurrence. The licensee states that common cause

failures of the sensors or the actuation logic is probabilistically insignificant and was not considered in developing the frequency of this event (RAI responses). In case of sensor failure, two low steam drum level and two low reactor water level signals would have to be generated. In case of actuator failure, two elements within one logic actuation cabinet would have to fail. It is not clear that this is probabilistically insignificant compared to the calculated IE frequency of 6.4E-5/yr, especially since this is substantially lower than the spurious depressurization frequency used in other BWR studies (orders of magnitude).

For event "spurious opening of turbine bypass valve", the data window was 27 years i.e., from 1966 through 1992, because the original bypass valve was replaced in 1965 due to unsatisfactory performance. Since that time there have been no events, but one event of spurious bypass closure at low power (<15% power) was counted in the calculation.

For many events where there have been no failure in the 29 year data window, one failure was assumed to have occurred and the frequency calculated correspondingly.

For the most part, and with the caveat about the RDS spurious actuation frequency above, the frequencies in Table 4 seem reasonable, except for the small small LOCA frequency which is substantially lower than the value found in NUREG/CR-4550, which was 2.E-2/yr (vs. the BRP value of 1.7E-3/yr).

It is stated that the LOCA frequencies used are up to an order of magnitude higher than those used in the 1981 study, which accounted for the fact that the plant had less piping than a typical nuclear power plant, due to its smaller size.

The LOCA and steam line break frequencies were derived from the EPRI BWR frequencies for small, medium and large breaks, and apportioned according to the BRP piping length fraction in different categories (4% for steam line breaks outside containment, 43% for steam line breaks inside containment, 11% for LOCAs above the core and 42% for LOCAs below the core). When a small small LOCA frequency is needed, the EPRI value for the small LOCA frequency is split evenly between the small and the small small LOCAs. The EPRI frequencies are conservative (by 2-3 orders of magnitude) when compared to the BWROG upper bound mean values which were derived using a "conservative statistical method of estimating break probabilities in a population in which no breaks have occurred". The BRP IPE states that the LOCA frequencies have been going down as more experience is gained with no breaks.

The RCP (recirculation pump) seal failure is not included in the initiating event frequency because procedures instruct the operators to trip the RCPs if the seal pressure and temperature cannot be maintained and because the RCP loop isolation valves could be closed to isolate excessive seal leakage if ac power is available.

The CDF at BRP is sensitive to the small-small LOCA frequency estimate.

It is stated in the submittal that NUREG/CR-4792 value of 1.E-3/yr/loop, for double ended guillotine break (DEGB) of SS-304 BWR recirculation piping is inappropriate for Big Rock Point for several reasons:

1) No credit is given for actions taken to mitigate IGSCC and no credit for inspections which detect cracks prior to failure;

2) The IGSCC mitigative actions have been demonstrated to be effective through an extensive testing program performed by EPRI/BWROG/GE in the 1980s, and have been recognized as effective by the NRC (NUREG-0313, Rev. 2);

3) The IGSCC analysis performed in NUREG/CR-4792 was performed primarily to compare the relative performance of 304 and 316 NG material, not to provide a point estimate.

It is also stated that BRP is not susceptible to IGSCC (except for two cases which occurred in the cleanup system due to specialized conditions associated with welds, which had been anticipated and have been corrected). The reasons for BRP "immunity" from IGSCC are the following:

1) The primary system is mostly cast stainless, specifically the 17 inch, 20 inch and 24 inch pipe.

2) The cast stainless has enough ferrite that microgranular cracks are arrested. It is generally agreed that cast stainless with 8% ferrite is immune to IGSCC, whereas selected BRP primary samples have shown the ferrite content to range from 10% to 25%.

3) The welding was done to minimize the thermal effects on the nearby heat affected zone.

4) BRP primary system has inherent flexibility (due to the configuration of the reactor vessel, the steam drum and the emergency condenser), which allows for thermal growth during power operation; this alleviates the residual stress, thus minimizing a necessary precursor condition for IGSCC.

For ISLOCA initiating event frequency, several pathways were considered (separate from the steam line break outside containment category) which bypass the containment. Valve failures were considered, but then a conditional probability of low pressure pipe rupture was also credited, given the high/low pressure isolation failure. It is assumed the pipe will fail due to internal pipe stresses and not because of the dynamic stresses due to a rapid pressurization. The conditional low pressure pipe rupture probability is 6.0E-3, based on BWROG data, as stated in the submittal. As the valve failure fault tree yields a frequency of high/low pressure boundary failure of 2.0E-5/yr for the core spray injection line, and 6.1E-3/yr for the fuel pit drain/blowdown line, then the initiating event frequencies for these two dominant pathways are 1.2E-7/yr for the former and 3.7E-6/yr for the latter.

Table 4 Big Rock Point Initiating Event Frequencies

Category	Initiator	Method*	Frequency(/yr)
LOCA, below core	Very small	2	1.7E-3
	Small	2	1.7E-3
	Medium	2	1.3E-4
	Large	2	3.0E-4
	RPV Rupture	2	2.7E-7
LOCA, above core	Small	2	8.8E-4
	Medium	2	3.3E-4
	Large	2	7.7E-5

Table 4 Big Rock Point Initiating Event Frequencies

Category	Initiator	Method*	Frequency(/yr)
Steamline break, inside containment	Very small	2	1.7E-3
	Small	2	1.7E-3
	Medium	2	1.3E-3
	Large	2	3.0E-4
Steamline break, outside containment	Small	2	3.2E-4
	Medium	2	1.2E-4
	Large	2	2.8E-7
General transients	Turbine trip	4	1.13
	Loss of feedwater	4	0.045
	Loss of main condenser	4	0.045
	Spurious MSIV closure	1	0.018
	Spurious bypass valve opening	4	0.049
	Spurious RDS valve opening	1,3	6.4E-5
Support system degradation	Load rejection	4	0.280
	Loss of DC power	3	0.045
	Loss of instrument air	4	0.045
ATWS	Turbine trip	2	1.1E-5
	Loss of feedwater	2	5.0E-7
	Loss of main condenser	2	4.5E-7
	Spurious bypass valve opening	2	4.9E-7

Table 4 Big Rock Point Initiating Event Frequencies

Category	Initiator	Method*	Frequency(/yr)
	Spurious MSIV closure	22	1.8E-7
	Loss of instrument air	2	4.5E-7
	Loss of Otisite power	2	6.1E-7
Special initiators	Interfacing system LOCA	1	3.8E-6
	Internal flooding	1	1.4E-6
Manual shutdown	Manual shutdown	4	5.60
	Loss of service water	4	0.09

*Method of calculating initiating event frequencies:
1. plant specific data
2. generic/industry data
3. system fault tree analysis
4. actual occurrences

2.2.3 Interface Issues

2.2.3.1 Front-End and Back-End Interfaces

Containment heat removal (i.e., the steam drum enclosure sprays) is modeled for certain initiators (steam line breaks inside containment) in order to insure survival of certain equipment inside the steam drum enclosure (e.g., emergency condenser valves, reactor level instrumentation, etc.). As mentioned above, at BRP injection can continue for substantial period of time following a LOCA, or another initiator, until the containment steel sphere design pressure (due to the hydrostatic head and the air pressure inside) is reached, at which point a switchover to recirculation [post incident system (PIS)] is attempted. In case PIS swithcover is unsuccessful, injection can be reinstated, such that the fill-the-ball strategy is used, wherein the containment is filled to its ultimate failure pressure. This strategy, while not credited, can buy substantial time, in addition to the long time periods expected for most accident evolutions.

It is claimed in the IPE that sufficient heat transfer can be obtained from cooling the submerged (or partially submerged) reactor vessel and through passive heat transfer to the containment structures and passive cooling of the containment shell (by the outside atmosphere), that no further injection or cooling may be necessary (although this is not credited in the analysis). The fact that the containment is a steel shell may have deleterious effects on certain operator actions (e.g., starting the standby diesel generator or aligning the portable pump) in case of radiation release inside the containment.

Section 2.4 provides more information on Level 2 considerations.

2.2.3.2 Human Factors Interfaces

The operator actions which may be important at this plant are: manual loading of the EDG or the SDG onto the emergency bus, manual starting of the SDG, alignment and starting of the portable diesel pump for emergency condenser makeup, manual depressurization or pressure control, actuation of the liquid poison system, recovery of the PIS (not credited) and fill-the-ball (not credited).

There are also some radiation concerns, mentioned above in 2.2.3.1.

Section 2.3 provides more information on HRA considerations.

2.2.4 Internal Flooding

2.2.4.1 Internal Flooding Methodology

The methodology used to perform the flooding analysis consisted of 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.

The development of flooding scenarios was supported by plant walkdowns.

The flood analysis was a refinement of previous flood analyses done for Big Rock Point to address certain issues, such as plant response to a break in circulating water piping following the Quad Cities 1 event of June 1972, leakage inside the containment and pipe breaks outside the containment. Some of these issues were addressed in the 1981 BRP PRA.

In this IPE analysis, several potential flooding areas were identified: the containment sphere, the turbine building, the core spray pump room, the screenhouse and the alternate shutdown panel building. Most of these areas were screened from further analysis based on qualitative arguments: the amount and flow rate of potential flood sources, alarms and other indications to the operator (e.g., startup of a fire pump on low system pressure), existence of a 2-hour operator patrol inside the containment, connection of the area to the outside, lack of sensitive equipment or a plant trip from a flood in the area. No estimate is provided of the CDF from the screened areas. All equipment in an affected area was assumed failed, both during the screening analysis and during the detailed analysis. It is noted that earlier studies have stated that certain equipment had splash guards, however, this is not credited here (RAI responses). However, if the area in question was open to other areas of the plant, such that there was no possibility of submergence, then no equipment in that area was failed, according to our understanding of the RAI responses. This is a non-conservatism, as spraying could still disable equipment. Some areas were screened out initially because the equipment had overhead splash guards, according to the submittal, thus contradicting the RAI responses. This would be an optimistic assumption if the spray could come from a direction other than directly overhead (no details were provided in the submittal or the RAI responses). No credit is given to floor drains, but possibility is considered of flood propagation through the drains, HVAC vents, etc. (RAI responses)

Mostly passive failures (pipe breaks) seem to have been analyzed. It is not clear whether maintenance induced floods were considered, other than to discount the maintenance errors committed at shutdown (RAI responses). However, maintenance errors at shutdown could, if undetected, cause a flood at full power.

In addition, potential flooding effects were part of the internal events analysis, where environmental effects seem to have been given a lot of attention, for example water level rising in a compartment from steam condensation and affecting the electrical equipment inside.

Only one flooding scenario survived the screening process. An event tree was constructed and quantified applicable to this scenario.

The possibility of isolating the flood seems to have been considered only qualitatively, as part of the screening analysis, but was not given credit in the detailed analysis of the remaining scenario.

In conclusion, the flooding analysis seems to have been appropriate, with some lingering questions about treatment of sprays and maintenance induced flooding.

2.2.4.2 Internal Flooding Results

The total CDF from flooding events is estimated to be 1.1E-9/yr. No estimate of the CDF due to the screened scenarios is offered in the IPE.

The one scenario that survived the screening was a flood in the screenhouse. This structure houses the condenser circulating water pumps, the service water pumps and the two fire pumps (the electric and the diesel driven fire pump).

A break in the piping of one of the systems is assumed to occur. The initiating event frequency for this event was calculated to be 1.4E-6/yr. This was calculated based on 182 ft of piping capable of flooding the screenhouse and the (high energy) pipe break frequency of 7.7E-9/ft/yr.

Assuming a flood flow rate of 500 gpm, electrical equipment in the downstairs portion of the screenhouse will begin to have contact with water in about 17 minutes. The 500 gpm is commensurate with the flow capacities of the various systems inside (1000 gpm for the fire pumps, 2100 gpm for the SW pumps, 24,500 gpm for the circulating water pumps, each).

Upon the postulated loss of the circulating water system due to the flood, the reactor and the turbine will trip due to loss of condenser vacuum. As the main condenser is unavailable, the desired response is to remove decay heat via the emergency condenser. As the firewater is unavailable, the shell side makeup to the emergency condenser can be accomplished by either the demineralized water system or the use of the portable diesel driven pump. In order to use the demineralized water option, and operate the air-operated control valves, alternate cooling (via domestic water) of the air compressors needs to be established, as the service water system is unavailable due to the flood. If emergency condenser long term cooling is established, it can be used until an SW pump is brought back into service, such that the shutdown cooling system can be placed in operation. Otherwise, the feedwater system can be used, in conjunction with gravity feed makeup to the condenser hotwell from the condensate storage tank (or from the demineralized water storage tank or from Lake Michigan), with pressure relief provided by the safety valves or the manual operation of the RDS system. Alternatively, CRD system can be used with manual operation of the RDS system and the hotwell makeup.

The dominant sequence in the above event tree involves a failure of the emergency condenser, failure of the feedwater, successful CRD operation but failure to manually control the reactor pressure using the RDS system. This sequence has a frequency of 1.1E-9/yr.

2.2.5 Core Damage Sequence Results

2.2.5.1 Dominant Core Damage Sequences

The results of the IPE analysis are in the form of systemic sequences, therefore NUREG-1335 screening criteria for reporting of such sequences are used. The point estimate for the core damage frequency from internal events is 5.4E-5/yr, with internal flooding contributing an additional 1.1E-9/yr. Accident classes and types and their percent contribution to the CDF, are listed in Tables 5a and 5b. The most important initiators are given in Table 6.

The dominant sequences were provided. Each of these important sequences has a frequency greater than 1.E-6/yr. The important sequences are summarized below in Table 7.

Table 5a Accident Classes and Their Contribution to the CDF

Accident Class	Contribution to CDF (/yr)	%
LOCAs with post incident recirculation failure	2.3E-5	42.6
LOCAs with core spray injection failure	1.3E-5	24.1
LOCAs with core damage at high pressure	7.5E-6	13.9
Transients with core damage at low pressure	4.5E-6	8.3
Transients with core damage at high pressure	2.0E-6	3.7
ATWS with failure of reactor inventory makeup	1.9E-6	3.5
ATWS with containment overpressure failure due to continued reactor makeup	1.9E-6	3.5
Station blackout	5.1E-7	0.9
Containment bypass	1.8E-8	0.03

Table 5b Initiating Event Categories and Their CDF Contribution

Initiating Event Category	Annual Frequency	%CDF
LOCA below core	3.2E-5	59.33
LOCA above core	7.7E-6	14.30
Support system transient	5.1E-6	9.44
ATWS	3.7E-6	6.96
SLB inside containment	3.4E-6	6.30
General transient	1.2E-6	2.18
Loss of offsite power group (load rejection, loss of station power, station blackout)	7.6E-7	1.42
SLB outside containment	2.5E-8	0.05
Other (ISLOCAs, flooding)	1.3E-8	0.03

Table 6 Dominant Initiating Events and Their Contribution to the CDF

Initiating Event	Contribution to CDF (/yr)	%
Very small LOCA below core	1.4E-5	25.27
Small LOCA below core	1.0E-5	19.50
Medium LOCA below core	6.3E-6	11.83
Loss of Instrument Air	4.8E-6	9.03
Small LOCA above core	4.6E-6	8.64
Turbine trip ATWS	3.4E-6	6.35
Medium LOCA above core	2.5E-6	4.66
Large LOCA below core	1.5E-6	2.73
Very small SLB inside containment	1.3E-6	2.50
Small SLB inside containment	1.1E-6	1.99
Manual shutdown	8.1E-7	1.51
Medium SLB inside containment	7.7E-7	1.44
Large LOCA above core	5.4E-7	1.00
Station Blackout	5.3E-7	0.99

Table 7 Dominant Core Damage Sequences

Initiating Event	Dominant Subsequent Failures in Sequence	% of CDF
Very small LOCA below core	long term failure of post incident recirculation	14.1
Small LOCA below core	long term failure of post incident recirculation	14.1
Medium LOCA below core	failure of core spray injection	7.4
Very small LOCA below core	failure of Reactor Depressurization System	6.5
Medium LOCA below core	failure of post incident system recirculation	4.3
Loss of instrument air	failure of emergency condenser makeup, successful depressurization, but loss of low pressure makeup with core spray or condensate	3.9
Turbine trip ATWS	successful turbine bypass, feedwater loss due to condensate reject to CST, liquid poison not injected before auto RDS; operator terminates core spray to avoid containment overpressure; core damage at low pressure with intact containment	3.2
Turbine trip ATWS	successful turbine bypass, feedwater loss due to condensate reject to CST, liquid poison not injected before auto RDS; operator fails to terminate core spray, reactor returns to power; containment fails on overpressure with core damage at low pressure	3.2
Small LOCA above core	failure of core spray and condensate makeup	3.2
Loss of instrument air	failure of emergency condenser initiation or makeup, failure of reactor depressurization	2.8
Small LOCA below core	failure of reactor depressurization	2.4
Small LOCA above core	failure of post incident recirculation	2.0
Medium LOCA above core	successful makeup with condensate, failure of post incident system	2.0
Loss of instrument air	failure of emergency condenser, SRV actuation leads to a stuck open SRV, failure of makeup with core spray and condensate	2.0
Very small LOCA below core	failure of core spray and condensate	1.9
Very small LOCA below core	failure of reactor depressurization	1.9
Small LOCA below core	failure of feedwater and core spray	1.9

The SBO contribution is about 1% of the CDF. The ATWS contribution is about 7%. The ISLOCA contribution is 0.024%, while all containment bypass sequences contribute 0.20% (including ISLOCA, steam line break outside containment and spurious bypass valve opening). System importances are calculated. The relative importance of systems seems reasonable for the design of the plant. Fussell-Vesely and Birnbaum importance measures are calculated. High Birnbaum importance indicates systems whose performance needs to be maintained in order not to substantially increase the CDF (i.e., CDF is sensitive to the data used for these systems). The systems which both have a high Fussell-Vesely and high Birnbaum importance are the critical systems. The most important systems from that standpoint are the firewater system, the core spray, the post incident, the reactor depressurization, the reactor protection system. These systems have a Fussel-Vesely importance greater than 0.10 and a Birnbaum importance greater than 1.0E-2/yr.

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 Big Rock Point 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. A broad range of both types of events were modeled, including 21 failure to restore events and 21 common cause miscalibration events. All pre-initiator or "latent" events were modeled in the fault trees.

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

All operator actions in the fault trees were identified by Big Rock Point analysts during the development of the logic models. Identification of the events were based on a review of each system and on plant procedures (operating and test and maintenance). While plant personnel were apparently involved in the identification and selection of pre-initiator actions, interviews with maintenance and instrumentation and control technicians regarding specific plant practices and application of procedures were not mentioned. However, the description of the application of the ASEP HRA procedure (NUREG/CR-4772) suggests that appropriate reviews of pre-initiator related practices and procedures did occur. Thus, it appears that relevant information sources were examined and that factors which could influence the probability of pre-initiator errors 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. Since the screening analysis in ASEP requires more or less the same activities as the nominal analysis, all restoration

events were given the detailed nominal analysis. In addition, common cause miscalibration errors were also given detailed analysis (no screening) using the THERP methodology (NUREG/CR-1278).

2.3.1.4 Quantification of Pre-Initiator Human Actions

As noted above, pre-initiator restoration faults were quantified using the ASEP method. The quantification process appeared to follow the method as documented and the resulting HEPs were reasonable. Plant-specific recoveries (PSFs) were appropriately considered, e.g., post-maintenance tests etc. Some fairly low HEPs (3.0E-6) were found in the tables documenting the analysis, e.g., Table 10-1, but these values reflected cases where indications regarding the restoration fault would be annunciated in the control room. The values appeared consistent with the methodology and many other licensees have simply not quantified restoration faults if appropriate indications are received in the control room. Dependence within a system was considered and quantified with guidance from the ASEP/THERP dependency tables. Possible dependencies between restoration errors that could affect components in more than one system were not quantified and this approach was appropriately justified on the basis of plant practices.

The common cause miscalibrations were calculated using a method based on THERP. In the response to the NRCs RAI, the Big Rock Point licensee provided examples of the calculations of common cause miscalibration HEPs and addressed why the range of HEPs was large, e.g., 7.5E-2 to 2.7E-6. Apparently some instruments are calibrated only using "data sheets" which documents the "as-found, as-left conditions" of the instrument, but do not provide detailed work instructions. The skill and knowledge of the I&C technicians are relied upon for these events. Thus, the resulting 7.5E-2 common cause HEPs. When detailed procedures must be followed, along with appropriate checks and sign-offs, lower HEPs are reasonably obtained. The general application of THERP appeared reasonable, as did the treatment of dependencies across similar instruments etc. While complete dependence across similar instruments was not assumed, at least some level of dependency was assumed and treated with appropriate THERP equations.

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 Big Rock Point IPE acknowledges both response and recovery type post-initiator human actions. However, post-initiator actions were modeled only when clear procedural guidance (normal, abnormal, or emergency procedures) existed for the operator and repair activities were apparently not credited. In addition,

the same methods were used to quantify all post-initiator actions. Thus, the distinction between the two types of events was not really relevant for the Big Rock Point IPE.

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

In the response to the NRCs RAI, the licensee indicates that there are some inaccuracies in section 4 of the submittal (method and approach). It is stated that section 4 was included to "address the reporting guidelines that required a concise description of the major tasks and methodology" and that "the intent from the BRP's PRA staff perspective was to provide an overview for plant personnel on how risk assessments were assembled." The licensee further states that "consequently several sections include *boilerplate*...that should have been edited after completion of the study." Given this assertion, it cannot be exactly determined how the post-initiator actions were identified and selected. Nevertheless, the process was apparently keyed to the plant response scenarios. Documentation from the submittal and the response to the RAI indicates that procedures were reviewed, appropriate personnel reviewed the models, interviews with operators and training personnel were held, and simulator exercises were conducted. The submittal also indicated that the "Control Room Design Review" was reviewed for human factors considerations and that a plant tour included locations outside the control room where operators actions would take place. Thus, reasonable steps were taken that would help ensure appropriate actions were identified and modeled.

In addition, it was also stated in the submittal that in most cases, "dynamic and recovery operator actions were not included in the fault tree and event tree models unless dictated by the models." Thus, many of the operator actions were included during "recovery" and could have been identified at that time.

2.3.2.3 Screening Process for Post-Initiator Response Actions

As noted above, in most cases, "dynamic and recovery operator actions were not included in the fault tree and event tree models." In addition, "all event tree sequences were solved with the initiating events set to 1.0 and a truncation limit of 1.0E-9" and " where it was initially recognized that resulting sequences may contain multiple operator actions, the HEPs were initially set to 1.0." After the initial quantification and when quantified operator actions were first included, the nominal ASEP HRA method was applied to all post-initiator human actions. Where important sequences contained multiple operator actions, the actions were analyzed to determine the dependencies between the HEPs. The HEPs obtained using the ASEP method are known to be somewhat conservative and from the licensees perspective, use of ASEP provided a fairly detailed screening approach. After the sequences were quantified with the ASEP values, operator actions identified as potentially being important were re-analyzed using the THERP methodology. All the actions re-analyzed had a Birnbaum importance greater than 1.0E-6. (The submittal notes that no latent human actions had a Birnbaum greater than this, so none of them were re-analyzed).

2.3.2.4 Quantification of Post-Initiator Human Actions

The application of the ASEP method to each event that did not receive quantification with THERP was documented in the submittal in Appendix C. The described derivation of the HEPs closely followed ASEP and seemed thorough and reasonable. Appendix C also documents the application of THERP to each of thirteen events identified as potentially important. While in many cases the application of THERP was also reasonable, there were several events for which the quantification process did not seem appropriate. It is thought that the resulting HEPs should be considered optimistic and that the use of such values for these events is a weakness of the HRA. The problem arises through the licensee's use of HEP values from the "annunciator response model" (Table 20-13 or Table 11-13 from THERP) in situations where very limited time (less than 10 minutes) is available for the operator action. While it can be argued that the HEPs from

this model are acceptable when substantial time (greater than 30 minutes) is available for the operators to determine the relevant actions and when the operators need only respond to the existence of an annunciator in the control room, the HEPs from this model do not reflect the impact of time on the likelihood of success. Thus, this model will underestimate HEPs for short time frame scenarios relative to the ASEP/THERP time-reliability diagnosis model.

The events of interest include the following:

1) Operators fail to restart the feedwater pump (FW-PM-P8SRT-POIC) after a trip in a LOCA scenario. Only 6 minutes are assumed available for diagnosis and the value used by the licensee for responding to the annunciator is 0.001, with the total HEP equal to 0.0046. The "best case" HEP from use of the diagnosis model would be at least 0.02, and with the action failures added in, the total HEP would be even higher. Moreover, there is little reason to assume that this is a "best case" event.

2) Operators fail to open MO-7073 & 7074 to provide make-up to the hotwell (FP-OO-MAKUP-POIC). This actions occurs in LOCA scenarios and also requires operators to locally open valve VFP-33 (this valve is apparently in a high radiation area of the plant). The licensee assumes that the actions only require 3 minutes (including the one outside the control room on top of the turbine shield) and that 6 minutes are available to diagnose/respond to the "low hotwell level and alarm." The total HEP for this event is listed at 0.016, but the diagnosis model, along with the fact that very little time is available for an ex-control room action in an area unshielded from containment, would be likely to produce a failure probability of 1.0 for this event.

3) Operators fail to trip the condensate pumps on low hotwell level (CD-HS-P9TRP-POIC). Two minutes are assumed available for this action during a small or medium LOCA and the total HEP is listed at 0.08. The diagnosis model would produce a diagnosis value alone of at least 0.5 and the total would be close to 1.0.

While the quantification of the above three events may be considered a weakness of the HRA, it cannot be concluded that the intent of the generic letter was not met in terms of identifying vulnerabilities related to the events. All three of the events were identified in the submittal as being relatively important in terms of contribution to CDF and a sensitivity analysis indicated that substantial increases in CDF would not be expected if the events were set to fail. Therefore, potential vulnerabilities related to these events were not overlooked. As noted above, the quantifcation of the remaining events was acceptable and as will be discussed below, other aspects of the HRA were generally done well.

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. In the licensee's discussion of the application of HRA methods, it appears that appropriate timing parameters were considered. The temporal occurrence of control room indicators were considered in determining the available time and in most instances (for exceptions see section 2.3.2.4. above) the impact of time on operator diagnosis was appropriately considered. Apparently MAAP runs were used to determine the latest time an operator action could be completed. General guidance from ASEP was used in determining other relevant times. In addition, the licensee states that walkdowns for important local actions occurred, taking into account timing, distances, environmental factors, required controls, location of indicators etc. Assumptions of available and required times were documented in the submittal.

2.3.2.4.2 Other Performance Shaping Factors Considered

PSFs addressed in the application of ASEP and THERP 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, and size of crew and time available for recovery credit were considered. These PSFs are those normally considered in applying the ASEP and THERP methodologies. In addition, the "control room design review" was reviewed for any additional human factors issues that should be considered and simulator exercises were observed to verify assumptions made during the HRA. A reasonable set of PSFs were apparently considered.

While the submittal states that environmental factors for local operator actions were considered, no explicit discussion of the potential impact of high radiation near equipment to be manipulated by operators was provided. Radiation could be a concern at Big Rock Point due to the lack of concrete shielding in containment. In response to a question on this topic in the NRCs RAI, the licensee indicated that no operator actions were credited after core damage. They then discussed three of the nine level 1 human action events which required actions outside the control. For one of the actions, which involved aligning the fire system for makeup to the hotwell, they note that a valve on top of the turbine shield has to opened and that "this area of the plant is not shielded from containment." The licensee then states that "as a result a very short time frame is conservatively assumed to complete this action." Presumably this statement means that the person performing this action will only be there for a short time. They clearly do not assume that there is time for an individual to put on protective clothing, but they do note that high stress was assumed for this event. Thus, the impact of radiation is apparently factored in to the HRA by assuming high stress. Additional information regarding specifics of a particular event would be needed to determine whether or not such treatment is adequate.

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 and THERP quantification approaches.

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. The Big Rock Point submittal indicates that evaluation of such timing factors occurred, but details were not provided.

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 indicate that potential dependencies among the operator actions were appropriately considered. In the initial analyses, multiple events in a sequence were set to 1.0. However, the submittal also states that except for two instances, sequences with multiple actions were only credited with one action (see page 10-14). All the remaining actions were left at 1.0

2.3.2.4.4 Quantification of Recovery Type Actions

As noted above, all post-initiator human actions were quantified using the same methods. Only actions directed by procedure were included and repair activities were not credited.

2.3.2.4.5 Human Actions in the Flooding Analysis

Eleven human actions were incorporated into the Big Rock Point flooding analysis (p. 7.1.6-11).

The eleven actions were also modeled in the overall internal events analysis, and while the flooding context etc. appeared to have been examined, the HEPs used were the same as those used in the overall internal events analysis.

2.3.2.4.6 Human Actions in the Level 2 Analysis

No operator actions were credited in the level 2 analysis.

2.3.2.5 Important Human Actions

The Big Rock Point IPE provided a list of important human actions as determined on the basis of a Fussel-Vesely and Birnbaum measures. Events identified with a Birnbaum greater than 1.0E-6 were included in the table in the submittal. The events, their Fussel-Vesely and Birnbaum values, and their HEPs are presented below in Table 8. As discussed above, the HEPs for at least three of the events listed must be considered optimistic.

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 18 key plant damage states (KPDSs). An event tree structure (called plant damage event tree in the IPE submittal), which includes containment conditions and containment system status as headings, is attached to the Level 1 event trees to determine the frequencies of the Plant damage states (PDSs) Based on their effects on Level 2 accident progression, these PDSs are then grouped to KPDSs to be used as the initiating events for containment event tree (CET) analysis. The containment parameters used in the IPE to define the PDSs include:

Containment bypass,
Containment leakage,
Enclosure spray system status,

Pool of water in containment, and
Inventory makeup available after vessel penetration.

The PDSs are defined by combining the containment states defined by the above parameters and the accident sequence subclasses (which are groups of Level 1 accident sequences). In the BRP IPE there are 16 accident sequence subclasses and 12 possible containment state definitions for each subclass. This yield a total of 192 PDSs, of which 83 are reported in the IPE with non-zero frequencies (response to RAI Level 2 Question 1). The 83 PDSs are grouped to 18 KPDSs for CET quantification.

The leading PDS (54% of total CDF) is represented by low pressure LOCA sequences with a pool of water in the containment before and after vessel penetration and the enclosure spray available. This is followed by a transient PDS with low reactor pressure, with a pool of water before vessel penetration, but with enclosure spray not available (10%), and another LOCA PDS with high reactor pressure (defined in the IPE as greater than 200 psig), with a pool of water and with the enclosure spray available (9%).

Other PDSs that are of interest to containment performance are those associated with containment failures. In the BRP IPE, containment integrity can be lost before vessel failure by overpressure, isolation failure, and bypass. PDSs which involve containment overpressure failure constitute about 4% of total CDF. The dominant PDS for this class is an ATWS PDS with both core injection and enclosure spray available.

PDSs with containment leakage contribute about 0.5 % to the total CDF. Containment leakage in the BRP IPE is a result of containment isolation failure. However, according to the IPE submittal, the main contributor to containment isolation failure for BRP is leakage as opposed to failure of isolation valve closure. Components such as the air locks and the vent valves make up the majority of containment leakage. The leakage probabilities used in the IPE quantification are determined from actual plant leak test data.

PDSs with containment bypass contribute about 1.5% to the total CDF. Containment bypass comes from either the level 1 bypass sequences or failure of the containment isolation valves that are connected directly to the reactor to close. Of the bypass PDSs, the contribution from the Level 1 bypass sequences is small (less than 0.1%). Although over 70% of the BRP sequences are LOCA sequences, the leading contributor to bypass PDSs is from transient sequences (about 1%). This, according to the IPE, is because the main steam isolation valve is required in some transient sequences to terminate the event, and failure to isolate the main steam line results in direct containment bypass.

In summary, for the BRP PDSs, the probability of successful containment isolation is about 94%, containment overpressure failure is about 4%, containment isolation failure is about 0.5%, and containment bypass is about 1.5%. The PDSs defined in the BRP IPE are of sufficient detail to provide a proper account of the front-end and back-end dependencies and adequate information for back-end accident progression analysis. Although the PDSs of various containment conditions are grouped to the same KPDS, the information is captured in CET quantification by CET headings for containment conditions.

Table 8 Important Human Actions

Event Description	F-V	Birnbaum	Human Error Probability (HEP)
Operator fails to line up RDS for pressure control using EIP-4 (RD-00-PCNTL-POIC)	?.0E-00	3.59E-05	1.50E-01
Operator fails to trip the condensate pumps on low hotwell level (CD-HS-P9TRP-POIC)	7.23E-02	4.84E-05	8.00E-02
Operator fails to trip the recirculation pumps during an ATWS and initiate LPS (LI-00-INJ2-POIC)	6.72E-02	?.20E-05	3.00E-00
Control Room operator fails to scram the reactor during a steam line break (RP-RX-VSSLB-POIC)	4.80E-02	4.28E-03	6.00E-04
Operator fails to open MO-7073 & 7074 to provide makeup to the hotwell (FP-00-MAKUP-POIC)	1.45E-02	4.84E-05	1.60E-02
Operator fails to align the post incident system, per SOP-8, following a LOCA (PI-OO-PISYS-POIC)	1.0E-02	6.47E-03	8.30E-05
Operator fails to manual open core spray valves during an SLB (CS-MV-CSVLV-POIC)	6.47E-03	5.58E-05	6.20E-03
Operator fails to restart feedwater pump (FW-PM-P8SRT-POIC)	5.89E-03	6.85E-05	4.60E-03
Operator fails to open the emergency condenser outlet valves (EC-MV-ECOUT-POIC)	5.07E-03	2.70E-05	3.70E-02
Operator fails to isolate TBV warm-up line on steam seal leak (MS-00-ISOLT-POIC)	2.65E-03	?.09E-05	1.30E-02
Operator fails to start SDG (standby diesel generator) from local control panel (EP-GE-SDG-POOC)	3.26E-04	?.98E-06	8.80E-03
Operator fails to trip the recirculation pumps during an ATWS and initiate LPS (LI-OO-INJ2-POIC)	2.87E-04 or 6.70E-02	?.54E-06	6.10E-02
Operator fails to provide make-up from fire protection system via SV-4947 (EM-KV-4947-POIC)	2.74E-04	?.73E-04	8.50E-05
Operator fails to back up automatic reactor scram (RP-RX-MANUL-POIC)	?.80E-05	?.50E-05	5.00E-03

2.4.1.2 Containment Event Tree Development

Probability quantification of severe accident progression is performed in the IPE by the use of containment event trees (CETs). The development of the CETs is discussed in Sections 12.5 of the IPE submittal. The CETs includes the following top events:

Key plant damage state (KPDS),
Containment bypass,
Containment leakage,
Early containment failure,
Enclosure spray (i.e., containment spray) available,
Ex-vessel debris coolability,
Late containment failure.

Figures 12.8-2 through 12.8-16 of the submittal show the CETs used in the IPE to determine the containment failure modes for the various KPDSs. It is noted that some of the parameters used in the CETs are also used in the definition of the PDSs (i.e., containment bypass, containment leakage, and enclosure spray status). They are included in the CETs because their status, although defined in the PDS, is lost in the KPDS in the binning process. In the BRP IPE, the binning of the PDSs to the KPDSs is based on their effect on containment accident progression, not on containment status. As a result, PDSs with various containment integrity and containment system status, but with similar effect on containment accident progression, are grouped to the same KPDS. Although the data related to these parameters are lost in the KPDS definition, they are recovered in CET quantification. The split fractions used in CET quantification for these parameters are obtained from the data obtained in PDS definition. In general, the CETs developed in the BRP IPE are well structured and easy to understand. The top events of the CET cover the important issues that determine the RCS integrity, containment response, and eventual release from the containment.

Fault trees are used in the IPE to quantify the top events of the CETs. The fault trees used in CET quantification are very detailed and address all phenomena and systems important to Level 2 accident progression. The quantification of the CETs is based on review of industry literature and plant-specific analyses using the MAAP-BRP code. 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 Bypass

Containment bypass is a PDS parameter. The results of the PDS event tree quantification are used to develop the event tree split fraction for this CET heading. In the BRP IPE, A very detailed fault tree structure is developed for the quantification of containment bypass. The BRP containment bypass fault tree includes those faults associated with an isolation failure of the process lines that connect to the primary system. The failure estimates of the basic events in the fault tree are determined by a combination of plant-specific and generic data. Results of the IPE analysis show that the main contributor to containment bypass failure for BRP is main steam isolation valve (MSIV) failure.

The failure of emergency condenser (EC) tubes due to high temperature creep rupture, which is considered in some other IPEs for BWRs with emergency condensers, is not considered a credible failure mode for BRP. This is based on the consideration of the EC tube design characteristics, the fact that both inlet and outlet EC

valves could isolate a failed tube, and that low RPV pressure events (i.e., LOCAs) constitute the largest fraction of the BRP core damage sequences (The licensee's response to RAI Level 2 Question 3).

Containment Leakage

Similar to containment bypass, containment leakage is also a PDS parameter. The results of the PDS event tree quantification are used to develop the event tree split fraction for this CET heading. In the BRP IPE, A very detailed fault tree structure is developed for the quantification of containment leakage. The BRP containment leakage fault tree includes those faults associated with an isolation failure of the penetrations that connect to the containment atmosphere. The failure estimates of the basic events in the fault tree are determined by a combination of plant-specific and generic data. Results of the IPE analysis show that door seals and vent valve leakage contribute over 90% to the total failure probability. For the containment access locks potential failures considered in the IPE include leakage past one door with the other door open or leakage past both closed doors.

In the PDS definition, containment overpressure failure (before vessel penetration) in an ATWS event is included in the containment leakage PDS. It is assumed in the IPE that failure to insert negative reactivity and continued RPV makeup will pressurize the containment to its pressure capability over the course of an hour.

Early Containment Failure

Early containment failure is defined in the BRP IPE as that which occurs at or shortly after vessel breach time. Key phenomena evaluated in the IPE include direct containment heating (DCH), energetic fuel/coolant interaction, and early hydrogen deflagration/detonation.

The probability of containment failure due to DCH is evaluated in the IPE by a decomposition event tree (DET) with the consideration of the RPV pressure at vessel failure, the containment pressure prior to vessel failure, the mode of RPV failure, the debris mass expelled from the RPV at vessel failure, the debris entrainment time, the fraction of debris fragmented and transported to the recirculation pump room, unoxidized Zirconium fraction in the debris, and peak containment pressure following RPV failure. The containment failure probability is then determined by comparing the containment pressure load with the containment fragility curve. According to the results presented in the IPE, containment failure probabilities for DCH vary from 2.3E-3 to 2.8E-1 for the various conditions described by the above parameters (Figure 12.4.1-1 of the submittal).

The potential failure modes for fuel coolant interaction considered in the IPE include those from (1) the transmission of a shock wave through water to the structure such as the reactor pedestal, (2) the impulse load from the shock wave through the containment air space, and (3) the loading by slugs of water propelled into containment structures. In-vessel steam explosions are considered highly improbable for BWRs, and are thus not considered in the BRP IPE.

For hydrogen, a two step approach is used in the IPE to investigate hydrogen concentrations in the BRP containment. The first is an integrated severe accident analysis using MAAP-BRP code, and the second step involves putting a bounding source of hydrogen into the enclosure room of the containment and evaluate its transport to other containment regions using the MAAP-BRP code. The study shows that the hydrogen concentration in all regions remain well below the detonation threshold. Early failure of the BRP containment due to hydrogen combustion is judged in the in the IPE to be of very low likelihood (~1E-3).

Based on the description provided in the BRP IPE submittal, all the important early containment failure modes discussed in NUREG-1335 are addressed in the IPE. Quantification of containment failure for the failure modes is based on data available in the literature and plant-specific results from MAAP-BRP code calculations. Although the probability of early containment failure from DCH can be high (e.g., 2.8E-2) under certain unlikely conditions, the probabilities of early containment failure from all early failure mechanisms vary from 0.002 to 0.006 in the BRP IPE (Figures 12.8-2 to 12.8-15). Considering the large containment volume to thermal power ratio and the strength of the containment, these values seem to be reasonable.

Debris Coolability and Late Containment Failure

In the BRP containment there is a 3 foot deep valve pit located in the center of the BRP CRD room, beneath the reactor vessel. This sump area has a cross section of 42 square feet and a volume of 126 cubic feet, and will collect core debris should a core damage accident progress to the point of vessel breach. A bounding calculation provided in the IPE submittal shows that the entire BRP core can be held within the CRD room sump with a total depth of 1.6 feet (50 cm). In the BRP IPE the probability of debris coolability (with water) is assigned a value of 0.5 for low pressure core melt scenarios and 0.9 for high pressure scenarios. However, It is assumed in the IPE that containment failure by basemat melt-through will not occur even if the debris is not coolable.

Debris coolability, as one of the sensitivity issues, is analyzed in the IPE by the MAAP-BRP code. The following phenomena that are related to debris coolability are discussed in the sensitivity analysis: non-condensable gas generation, debris cooling in the sump, concrete attack, and containment debris spreading. According to the sensitivity analysis, concrete attack depth in "dry" cases, or in cases with very limited debris to water heat transfer coefficients, would be unlikely to lead to basemat failure in any reasonable length of time (e.g., a 36 hour mission time used in the BRP IPE). For BRP there is approximately 10 feet of concrete below the CRD room sump.

Late containment failure is a CET top event. The fault tree used in the IPE to determine the probability of late containment failure include two basic events: One involves long term high containment pressure failure and the other involves long term high containment temperature failure. According to the containment event trees (Figures 12.8-2 to 12.8-15), the probability of late containment failure is assigned a value of 1E-4 for all KPDSs. The probability of late containment failure for BRP is small because of the large containment volume and the use of a 36 hours mission time (licensee's response to RAI Level 2 Question 9).

In-Vessel Recovery and Fill-the-Ball Procedure

In the BRP EOPs, "Fill-the-Ball" is used to provide water to the containment structure for reactor vessel heat removal in the event of a post incident system (for long term heat removal) failure. The fill-the-ball strategy will incorporate any available makeup system to provide continued water supply for heat removal. This will result in the submergence of the reactor vessel and cooling of the core debris in the lower head of the vessel. Vessel failure may be prevented by this cooling mechanism. However, because of the uncertainties associated with this cooling method, it is not credited in the IPE as being able to provide adequate core cooling to prevent vessel failure. Although failure is always assumed to occur it is incorporated into the LOCA event trees in the IPE as a potential mitigative action. Its effect on containment pressure, temperature and environmental fission product release is investigated in the IPE in the sensitivity analysis.

2.4.1.3 Containment Failure Modes and Timing

The BRP containment ultimate strength evaluation is described in Section 12.3 of the IPE submittal. The ultimate containment failure pressure for the BRP IPE is estimated by plant-specific analysis and results are compared with that obtained in the IDCOR report for the GE Standard Mark III containments. The median pressure capacity obtained in the BRP IPE (79 psig) is the average between the service Level C pressure and the pressure based on the ultimate tensile stress. The containment failure pressure distribution (i.e., the fragility curve) is assumed to be a normal distribution with a coefficient of variation of 13 psi. This is based on the consideration of both the uncertainties involved in material strength and modeling. The derivation of the containment fragility curve for BRP seems to be adequate and the results seem to be consistent with those obtained in other IPEs.

2.4.1.4 Containment Isolation Failure

Two failure modes are considered in the BRP IPE for the containment isolation systems - bypass and leakage. Fault trees are used in the IPE to evaluate the probabilities of these two failure modes. The containment bypass tree include those faults associated with an isolation failure of the process lines that connect to the primary system, and the containment leakage tree deals with the penetrations that connect to the containment atmosphere. According to the descriptions provided in the IPE submittal and the licensee's response to the RAI (Level 2 Question 5), 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

No recovery actions are credited in the Level 2 analysis of the BRP IPE.

2.4.1.6 Radionuclide Release Characterization

Radionuclide release categories are discussed in Section 12.6 of the IPE submittal. The CET end states are grouped to release categories based on the following parameters:

> Timing of release,
> Early -- Less than 6 hours from accident initiation,
> Intermediate -- Greater than or equal to 6 hours, but less than 24 hours,
> Late -- Greater than or equal to 24 hours,
> Total quantity of fission products released,
> High -- A release of sufficient magnitude to cause near-term health effects, greater than 20% CsI release fraction,
> Moderate -- A release with the potential for latent health effects, between 10% to 20% CsI release fraction,
> Moderate-Low -- A release with minor health effects, between 1% to 10% CsI release fraction,
> Low -- A release with the potential for minor health effects,, between 0.1% to 1% CsI release fraction,
> Low-Low -- A release that is less than or equal to containment design base leakage resulting in no health effects, less than 0.1% CsI release fraction.

According to the IPE submittal, the definition of release timing is based upon past experience with offsite responses. Emergency Action Level is considered in the definition. The time to the declaration of a General Emergency is estimated in the IPE to be about 1 hour or less for most accidents.

The classification of the release magnitude is based on the review of existing consequence analyses performed in previous IDCOR studies, PRAs, and NRC studies containing detailed consequence modeling. Based on the review, the release fraction of CsI is used in the IPE for release magnitude classification. It is used because it correlates well with the predicted latent effect and shows a threshold value for predicted early fatalities in previous consequence analyses.

The use of the above classification method results in fifteen source term release categories. A series of MAAP-BRP calculations were performed in the BRP IPE to assign the proper source term category to the CET end states. The MAAP-BRP calculations usually terminated at 36 hours after vessel failure. Since containment failure in BRP is primarily due to bypass or leakage prior to vessel failure, which, according to MAAP-BRP calculations, occurs before 6 hours, the release timing for the containment failure cases for BRP is primarily early.

Although there are 15 possible release categories, the CET quantification results show only 5 release categories with non-zero frequencies (Figure 12.8-17 of the submittal). Besides a no containment failure category, all the other release categories involve early releases. The conditional probability for the no-containment-failure category is about 94%. The next release category, which contributes about 4% to total CDF, has a CsI release fraction 0.1% to 1%. It is primarily from ATWS and LOCA sequences with enclosure spray available. The release category that has high CsI release (i.e., greater than 20% release fraction) contributes about 1.5% to the total CDF. It is mostly from containment bypass sequences in which the enclosure spray and any water collected in the sump provide no benefit in limiting the source term severity.

Except for the release fractions of CsI, which is used in the IPE for source term classification, the release fractions of other fission product categories (e.g., Te and Sr), are not provided in the IPE submittal. Additionally, only ranges of Cs releases (e.g., between 0.1% to 1%) are provided in the IPE submittal for the various source terms. This seems to be sufficient to provide a general characterization of the BRP containment performance in a severe accident, but is not sufficient for a detailed consequence analysis. Since MAAP-BRP code calculations were performed in the IPE for selected sequences, the release fractions for other fission products categories, although not reported in the submittal, were available from the MAAP-BRP calculation results.

The discussion of source term classification provided in the IPE submittal is detailed and reasonable. The grouping of the CET end states to source term release categories also seems reasonable. However, besides CsI, the release fractions of other radionuclide groups are not reported in the IPE submittal.

2.4.2 Accident Progression and Containment Performance Analysis

2.4.2.1 Severe Accident Progression

In the BRP IPE, a modified version of the MAAP 3.0B code (MAAP-BRP) was used in evaluating the reactor pressure vessel (RPV) and containment responses and determining the resulting source term. MAAP-BRP is based on the version of the MAAP code developed by the Department of Energy's Advanced Reactor Severe Accident Program, in corporation with General Electric, for the GE Simplified BWR (SBWR). Table 12.7.1-1 of the IPE submittal shows the MAAP-BRP calculation results for 29 cases. The key plant conditions considered in the selection of the MAAP-BRP cases include: containment bypass, containment isolation, early failure of containment due to energetic event at vessel failure, containment sprays, ex-vessel debris coolability, late containment failure, and reactor pressure.

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

Radionuclide release categories (or containment failure modes) and their frequencies obtained from the BRP CET quantification are discussed in Section 12.8 of the submittal. Table 9, below, shows a comparison of the conditional probabilities for the various containment failure modes obtained from the BRP IPE with those obtained from the Surry and Zion NUREG-1150 analyses.

Table 9 Containment Failure as a Percentage of Total CDF

Containment Failure Mode	BRP IPE++	Surry NUREG-1150	Zion NUREG-1150
Early Failure	4.2	0.7	1.4
Late Failure	+++	5.9	24.0
Bypass	1.5	12.2	0.7
Isolation Failure	***	*	**
Intact	94.3	81.2	73.0
CDF (1/ry)	1.7E-5	4.0E-5	3.4E-4

++ The data presented for BRP are based on Figure 12.8-17 of the IPE submittal.

+++ Late containment failure is assigned a probability of 1E-4 in the CETs presented in the IPE submittal. However, results presented in Figure 12.8-17 shows a zero probability for late failure. The negligible late failure probability is due to the large containment volume and the use of a 36 hour (after vessel failure) mission time.

* Included in Early Failure, approximately 0.02%.

** Included in Early Failure, approximately 0.5%.

*** Included in Early Failure. Of the 4.2% probability of early failure about 0.5% is from leaks through penetrations.

As shown in the above table, the conditional probability of containment bypass for BRP is 1.5% of total CDF. Of the 1.5% bypass probability, only 0.06% comes from Level 1 bypass sequences (i.e., ISLOCA), and the majority is from failure to isolated the process lines that connect to the primary system. Although LOCA is the dominant contributor to total plant CDF, the main contributors to this failure mode are transients (0.8% of CDF). This is because MSIV closure is required in some transient sequences to terminate the event, and failure to isolate the MSIV results in direct containment bypass.

The conditional probability of early containment failure for BRP is about 4.2% (of total CDF). The leading contributor to this failure mode is containment overpressure failure before vessel breach in ATWS events (3.5% CDF). This is followed by leakage through containment penetrations (0.5%), mostly from leakage through vent valves and door seals. Containment penetration leakage comes primarily from LOCA (about 80% of leakage cases) and transient sequences (about 20% of leakage cases).

Because of the large containment volume and the use of a 36 hour mission time, containment failure by the energetic events at vessel breach and long-term pressurization and thermal attack is not likely at BRP.

2.4.2.3 Characterization of Containment Performance

As shown in Table 9, for Big Rock Point Plant, the core damage frequency (CDF) is lower than that obtained in NUREG-1150 for Zion and Surry. Although the containment volume to thermal power ratio is much larger for BRP than for Zion and Surry, the conditional probability of early containment failure for BRP is greater than that for Zion and Surry. The major contributor to early containment failure for BRP is not from energetic event associated with HPME or hydrogen burn but from the steam generated during ATWS events and leakage through containment penetrations. This may be partially attributed to the use of a BWR (for a more severe ATWS load) and the accessibility of the containment during normal operation for BRP (for more severe leakage cases).

The mechanisms of containment bypass for BRP is also different than those for PWRs such as Zion and Surry. SGTR, which is an important bypass failure mode for PWRs, is not important for BRP[2]. The major contributor to containment bypass in BRP is the failure to isolated the processing lines connected to the primary system, primarily the lack of isolation of the MSIV.

The C-Matrix, which shows the conditional probabilities of release categories (or containment failure modes) for the plant damage states (or KPDSs), can be obtained from the data presented in Figures 12.8-18 and 12.8-19 of the submittal.

2.4.2.4 Impact on Equipment Behavior

The enclosure spray is a system that is considered in the CET quantification. Containment fan coolers that are not intended to perform a safety function and are isolated upon a containment isolation signal at BRP they are not credited in the IPE. The availability of enclosure spray is determined in PDS definition and its availability in CET quantification is primarily based on PDS definition. As a result, the effects of harsh environmental conditions on the operation of enclosure spray are not addressed in CET quantification. This is not a serious problem because the probability of long term containment failure is negligible. The primary effect of the enclosure spray in the BRP IPE is the scrubbing of fission products in the early failure cases. Since the spray pumps are located outside the containment the effects of harsh environmental conditions on its operation do not seem important.

2.4.2.5 Uncertainties and Sensitivity Analysis

Sensitivity studies for Level 2 analyses are discussed in Section 13.7 of the IPE submittal. The uncertainties were, in general, addressed quantitatively in the BRP IPE, using deterministic methods, and MAAP-BRP was selected to perform plant specific evaluations. The sensitivity studies provided in the IPE submittal address the uncertainties associated with the following phenomena:

Core melt progression -- amount of residual debris in RPV,
In-vessel hydrogen generation -- core blockage,
RPV pressure at vessel failure,
Debris coolability,

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[2] A failure mode in BRP that is similar to that of SGTR for PWRs is that associated with creep rupture of the emergency condenser tubes. It is not important for BRP.

Containment failure mode (size, location, and type), and
Containment flooding sensitivity.

The selection of the above issues for sensitivity analyses is based on the consideration of the sensitivity issues raised by NRC and the industry (e.g., the EPRI recommendations).

As discussed above, the sensitivity studies performed in the BRP IPE are deterministic sensitivity studies. The sensitivity studies were performed by varying some MAAP parameter values from their base case values and analyzing the differences in MAAP calculation results. The effects of uncertainties on CET quantification results are not addressed directly. For example, it is not clear (from the sensitivity studies presented in the IPE submittal) what is the effect of the amount of core forced out of the vessel (which is one sensitivity study item, Section 13.7.3.1) on DCH load, and consequently, the probability of early containment failure. Although such probabilistic sensitivity studies were not discussed in Section 13.7 of the submittal, the uncertainties on some key containment phenomenological issues were discussed in some detail in CET quantification. For example, early containment failure due to DCH was evaluated in the IPE by the use of a decomposition event tree (DET). The top events of the DET addressed the issues of significant uncertainties for DCH. Results of the DET analyses presented in the IPE submittal show that the probabilities of early containment failure due to DCH vary from 0.002 to 0.28 (Figure 12.4.1-1). The high value (i.e., 0.28) obtained in the IPE seems to be a conservative estimate and not likely to occur because of the large containment volume to thermal power ratio and the strength of the containment.

Although probabilistic sensitivity analyses are not discussed specifically in the sections addressing sensitivity issues in the IPE submittal, the accident phenomena that have significant uncertainties on containment performance are evaluated and discussed in detail in the IPE submittal. The sensitivity studies provided in the BRP IPE seems to have addressed the issues of significant uncertainties in the IPE analysis.

2.5 Evaluation of Decay Heat Removal and Other Safety Issues

This section of the report summarizes the review of the evaluation of Decay Heat Removal (DHR) provided in the submittal. Other GSIs/USIs addressed in the submittal were also reviewed.

2.5.1 Evaluation of Decay Heat Removal

2.5.1.1 Examination of DHR

The IPE addresses decay heat removal (DHR). Several systems performing the DHR function are mentioned including main condenser, feedwater, emergency condenser, reactor depressurization, low pressure injection with the condensate system, low pressure injection with the core sprays, long term recirculation with the post incident system and containment flooding (or fill-the-ball). CDF fractions were estimated in which these systems had failed, as follows: emergency condenser (2.9E-3), emergency condenser firewater makeup 2.6%, reactor depressurization (14%), main condenser (7.4E-4), post incident system (17%), core spray valves (40%), feedwater system (4.8E-3), fill-the-ball (42%), condensate system (3.8E-3).

The unavailability of each system is shown for important initiators. Support systems are identified. Contribution of important hardware failures and important operator actions associated with the system are calculated (Fussel-Vesely importance).

Several human actions are identified as being important to meeting the DHR capability. The actions include latent (pre-initiators) and post-initiator actions. They include actions related to the availability of the main condenser, feedwater, the emergency condenser, condensate, and the post-incident system.

The licensee considers this issue closed.

2.5.1.2 Diverse Means of DHR

The IPE evaluated the diverse means for DHR, as shown in Section 2.5.1.1 above.

2.5.1.3 Unique Features of DHR

Section 1.2 of this TER describes unique plant features, most of which are DHR features

2.5.2 Other GSIs/USIs Addressed in the Submittal

In addition to USI A-45 (DHR Evaluation) the following USIs and GIs are considered closed by the licensee as a result of the IPE submittal:

1) USI-A43, "Containment Sump Emergency Performance", which deals with post LOCA flow blockage (see discussion in Section 1.2), and

2) Closure of BRP Severe Accident Management Guidelines.

In addition, the IPE will be used to comply with the maintenance rule.

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 are discussed in the submittal, the CPI issue is not specifically addressed. More detailed information on this issue is provided in the licensee's response to the RAI (Level 2 Question 2).

According to the response, several containment walkdowns were conducted to support the use of a Generalized Containment Model (GCM) which allows the evaluation of local effect and requires detailed containment information. Personnel involved in the walkdowns included the BRP Plant Manager, PRA staff members, and one of the GCM code authors. The walkdowns focused on all the different modes of room interaction ranging from hydrogen to debris transport. The PRA staff continues to monitor the containment status through video taping and walkdowns subsequent to the IPE submittal.

The evaluation of hydrogen concentrations and combustion in the BRP containment showed that because of the large containment volume, the hydrogen concentration in the BRP containment would only be 10% if 100% of Zircalloy is oxidized and all the steam is condensed. Potential hydrogen release locations were also examined in the IPE and showed that high localized hydrogen concentration is not likely to occur and that the BRP containment will be well mixed. Hydrogen detonations are not credible for the BRP containment and the pressure loads from hydrogen deflagration are not likely to challenge the integrity of the containment.

2.6 Vulnerabilities and Plant Improvements

According to the licensee, a vulnerability would be identified by answering the following questions (RAI responses, page 5 and 6 of RAI responses, question 4 from the January 29, 1996 NRC request for additional information):

1) Are there any new or unusual means of reaching a situation in which core damage or containment failure would occur that had not been identified in previous PRAs?

2) Do the results of the PRA suggest that the Big Rock Point Plant would contribute to the industry not meeting published safety goals?

Based on the above definition, no vulnerabilities were found. It is stated that when compared to the newer facilities, the large and robust containment is at least as effective (if not more) in limiting the potential for significant releases following a severe accident.

No plant modifications were planned as a result of the IPE. A list of plant improvements following the 1981 PRA is provided in the RAI responses. These improvements have already been implemented.

No SBO rule related hardware changes have been implemented. Some procedural changes in response to the SBO rule have been implemented.

Several human action related improvements were discussed in the licensee's response to the RAI. Not all of these improvements were directly related to the results of the PRA, but were related to the process of performing the IPE. The improvements included:

- Eight manual valves in the post incident system were modified such that they cannot be locked except in their correct position.

- A contingency was provided to EOP EIP-5 to allow operators to implement containment flooding as a Severe Accident Management (SAM) strategy.

- The "Control Room Design Review" performed at Big Rock Point took advantage of the dominant accident sequences from the 1981 PRA to perform its task analysis and resolve potential human error deficiencies.

- The complement of the PRA developed fault trees was used in writing the FORTRAN logic handlers for the simulator at Big Rock Point. As a consequence, the need to manually trip the condensate pumps after the hotwell had emptied was identified.

No other plant improvements were discussed in the IPE submittal.

3 CONTRACTOR OBSERVATIONS AND CONCLUSIONS

Strengths of the IPE Level 1 analysis are as follows: Thorough analysis of initiating events and their impact, descriptions of the plant responses, modeling of accident scenarios, reasonable failure data and common cause factors employed and usage of plant specific data where possible to support the quantification of initiating events and component unavailabilities. The treatment of dependencies and environmental effects seems very thorough, as does the sensitivity and importance analysis. The effort seems to have been evenly distributed across the various areas of the analysis. The documentation is very detailed, and there seems to have been a conscious effort to respond to the RAIs to the best of the licensees ability.

There are some areas of concern related to the IPE but these are not expected to have a major impact on the conclusions. In the area of initiating event frequency, the RDS spurious opening frequency and the small LOCA frequency seem low. As far as data is concerned, data for the fire pumps seem low (corrected by a sensitivity analysis). There are questions as to the Bayesian updating algorithm used, as unreasonable values are produced for the fire pumps (the other components seem OK). The common cause failure between the electric and diesel driven pumps is not considered; the common cause failure between the two station batteries is not considered. Also, the system Birnbaum importance calculation algorithm seems to break down at high Birnbaum importance values (partially corrected by recalculating systems which seem to have a problem, per RAI responses) (corrected Birnbaum importance values are quoted in this technical evaluation report). There is a question as to why only pipe failures seem to be important in the flooding analysis. It is not clear if maintenance induced floods and spray effects were treated properly. Finally, the documentation is sometimes self-contradictory, and some part of it seem not to have been reviewed prior to publication.

The IPE determined that LOCAs contribute about 80% to the CDF at BRP. The most important sequences have failures of the post incident system, the reactor depressurization system and/or the core spray system. The interfacing system LOCAs and containment bypass sequences show negligible contribution to the CDF. The same can be said for the flooding scenarios. The blackout contribution is small (1%), due to existence of the 100% load rejection capability, the emergency condenser, the ac independent makeup to the emergency condenser and long life of the alternate shutdown battery, as well as existence of two diesel generators (albeit with limited capability). The loss of instrument air contribution is relatively large (9%) due to its usage for emergency condenser makeup from demineralized water, feedwater flow control, feedwater pump cooling, and main condenser hotwell makeup. The ATWS contribution (7%) is governed by two opposing forces: less time than at other BWRs is available for injection of the standby liquid control system, due to nonexistence of a high pressure high volume ECCS system at BRP; however the SLCS at BRP is a fast acting one that ensures subcriticality in about 1 min after operator actuation. The 100% bypass capability is not credited as the operators have to trip the recirculation pumps in a very short time in order to avoid losing the feedwater system (even though they were able to accomplish this in training exercises). Also, the trip frequency at BRP seems to be higher than at other plants.

The BRP Level 1 risk profile does not look like that of a typical BWR, where blackout and ATWS usually dominate the core damage frequency. Here LOCAs dominate, with ATWS contributing (in the absolute sense) about the same or slightly higher CDF than most other BWRs due to the features mentioned above. The blackout contribution is much smaller than at other BWRs, as explained above. There are several reasons for the high LOCA contribution: a portion of primary piping is located below the level of the core, which leads to a more severe case of LOCAs; there is paucity of high pressure (/high flow rate) makeup systems; for larger LOCAs, makeup to the condenser hotwell is inadequate which leaves the two fire pumps as the only low pressure system available; some important systems would be disabled by the harsh environments due to LOCAs and/or steam line breaks; lack of suppression pool means that at some point recirculation must be

brought into play (called the post incident system); and finally, no credit is given for fill-the-ball proceduralized action, with its passive cooling features, if recirculation fails. On the other hand, it is stated that the BRP piping is not subject to the inter-granular stress corrosion cracking (IGSCC) which plagues other BWRs; the recirculation strainers are much less vulnerable to plugging than at other BWRs; it is also claimed that the LOCA initiating event frequencies are conservative.

The HRA review of the Big Rock Point 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:

1) The submittal indicated that utility personnel were involved in the HRA. Procedure reviews, discussions with operations and training staff, observations of simulator exercises, review of the "control room design review", and walkdowns of important operator actions, including local actions, helped assure that the IPE HRA represented the as-built, as-operated plant.

2) The HRA process for the Big Rock Point 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. All pre-initiator restoration errors were analyzed in detail (no screening analysis) and quantified using the ASEP HRA procedure (NUREG/CR-4772). All common cause miscalibrations were quantified in detail using a method derived from THERP (NUREG/CR-1278). A reasonable and thorough analysis of pre-initiator events was performed.

3) In general, the licensee's analysis of post-initiator events was performed reasonably. A detailed "screening" was performed and important human actions were given an even more detailed analysis. However, there were several events for which the quantification process did not seem appropriate. It is thought that the resulting HEPs should be considered optimistic and that the use of such values for these events is a weakness of the HRA. The problem arises through the licensee's use of HEP values from the "annunciator response model" (Table 20-13 or Table 11-13 from THERP) in situations where very limited time is available for the operator action. While it can be argued that the HEPs from this model are acceptable when substantial time is available for the operators to determine the relevant actions and when the operators need only respond to the existence of an annunciator in the control room, the HEPs from this model do not reflect the impact of the time available on the likelihood of success. Thus, this model will clearly underestimate HEPs for short time frame scenarios and the ASEP/THERP time-reliability diagnosis model is clearly indicated in such situations.

4) Plant-specific performance shaping factors (PSFs), event timing, and dependencies were apparently appropriately considered in most instances. However, for the three events discussed above, timing was not considered appropriately. In addition, in at least one event the licensee *may* not have appropriately factored in the impact of potential radiation hazard on operator performance.

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

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

The interface between the Level 1 and Level 2 analyses is accomplished by the development of a set of 18 key plant damage states (KPDSs). The Level 1 core damage sequences are grouped in the KPDSs based on accident types (e.g., transient. LOCA, ATWS, ISLOCA), RCS pressure, and the availability of injection systems. The CET used in the BRP IPE include 7 top events addressing containment and containment system conditions, various modes of containment failure, and mechanisms that affect fission product release. CET quantification is based on the data available in the industry literature and plant-specific analysis using MAAP-BRP code. The definition of the PDSs 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 important points of the technical evaluation of the BRP IPE back-end analysis are summarized below:

1) 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.

2) The Big Rock Point Plant IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.

4 REFERENCES

[IPE] Big Rock Point Plant Individual Plant Examination, October 11, 1996.

[RAI Responses] Response to NRC Request for Additional Information, Big Rock Point Plant IPE

[NUREG/CR-1278] A.D. Swain and H.E. Guttman, *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Applications : Technique for Human Error Rate Prediction*, NUREG/CR-1278, U.S. Nuclear Regulatory Commission, Washington D.C., 1983.

[NUREG/CR-4772] A.D. Swain, *Accident Sequence Evaluation Program Human Reliability Analysis Procedure*, NUREG/CR-4772, U.S. Nuclear Regulatory Commission, Washington, D.C., February, 1987.

[NUREG-1150] USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.