TECHNICAL REPORT
FIN W-6449 2/23/96

TECHNICAL EVALUATION REPORT OF THE IPE SUBMITTAL AND RAI RESPONSES FOR THE INDIAN POINT NGS, UNIT 2

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Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Contract No. DE-AC02-76CH00016

Attachment

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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 Indian Point Nuclear Generating Station, Unit 2. The primary intent of the review is to ascertain whether or not, and to what extent, the back-end IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 and achieves the four IPE sub-objectives. The review utilized both the information provided in the IPE submittal and additional information (RAI Responses) provided by the licensee, the Consolidated Edison Company of New York, in the response to a request for additional information (RAI) by the NRC.

E.1 Plant Characterization

The Indian Point 2 Nuclear Power Plant is a Westinghouse 4 loop pressurized water reactor (PWR). The plant has a power rating of 3083 Mwt (the plant power rating has been increased by 11.5%). IP-2 is operated by Consolidated Edison Company of New York.

Collocated on the site are Indian Point Unit 3 (3,025 Mwt), operated by the New York Power Authority, and the older, decommissioned Unit 1. No mention is made in the submittal of any dual unit effects between Units 2 and 3 and responses to the RAI indicate that Indian Point 2 and Indian Point 3 are basically independent of each other. Some auxiliary systems from Unit 1 can be used in an emergency. The Indian Point site is located on the east bank of the Hudson River, in Westchester County, New York.

A number of design features at Indian Point 2 impact the core damage frequency. However, the submittal does not highlight all of these features nor does it estimate their effect on the core damage frequency. The following features tend to decrease the CDF:

- **Feed and bleed capability.** This capability is not singled out in the submittal as being important, however, it is credited in the event trees and is proceduralized, according to the IPE. However, the time available is somewhat limited, and the plant is considering changing the EOPs to require feed and bleed initiation at an earlier point in time.

- **City water supply connection to the AFW system.** The 1.5 million gallon City Water Storage Tank can be aligned to the AFW (auxiliary feedwater system) pumps should the 360,000 gallon can condensate storage tank (CST) become unavailable or exhausted. This is not mentioned in the IPE as being an important design feature.

- **Manual AFW control following station blackout.** The design of the turbine driven AFW pump (TDAFW) permits operators to manually control it even when DC power has failed (in a prolonged blackout). Pneumatic instrumentation for the pressurizer and the steam generator is also available. Ventilation and cooling to the area housing the AFW equipment can be provided by opening the roll up door which gives direct access to the transformer yard. The IPE mentions this manual AFW control as an important design feature.

- **City water/primary water cooling of HPI, RHR and charging pumps.** The normal source of cooling for the RHR, SIS (i.e., HPI) and charging pumps is the component cooling water system. However, in the event of a loss of component cooling, a hard-piped city water pump cooling connection to the charging pumps is available, and the procedure specifically directs the operator to make the proper alignment. At the same time, the procedure directs the operator to align the HPI and the RHR pump cooling to either the city water or the hard-piped primary makeup water connection. In addition, the RHR (i.e. the LPI) and the HPI pumps may be operated in the injection mode without cooling, whereas the charging pumps can be operated at maximum speed to extend the time they may be operated without cooling.

  Thus the potential for an RCP seal LOCA, or a LOCA associated with a loss of CCW or non-essential service water, is significantly diminished.

- **Emergency power system.** IP-2 has three emergency diesel generators plus three gas turbines, which can be used if the diesel generators fail. Gas turbines 1 and 3 have full black start capability. Gas turbine 1 is located on the IP-2 site, whereas gas turbines 2 and 3 are located close by at the Buchanan substation. The operating procedures direct the operators to start up and connect the gas turbines in case of an accident. The plant will be adding the blackstart capability to gas turbine 2 also.

- **Recirculation capability.** Two sets of low pressure pumps are provided for recirculation: two RHR pumps(outside the containment) and two LPR pumps (inside the containment). The water is recirculated from the sump, through the RHR heat exchangers and into the RCS and/or the spray headers for containment pressure control.

- **Service water.** The essential and the non-essential service water headers may be aligned to different service water pumps in case of pump failure (each SW system has 3 pumps). This is not mentioned in the IPE as an important feature.

- **High to Low pressure piping interface.** Since the high to low pressure interface on the RHR system is inside the containment, there is a strong possibility that should an interfacing system LOCA occur due to failure of suction valves, it will occur inside the containment. This significantly reduces the frequency of containment bypass events. For the dominant interfacing LOCA sequence, which involves failure in the RHR suction piping, neither the submittal nor the RAI responses specify what fraction of the piping is inside the containment and how the probability of piping failing outside the containment is calculated.

- **Containment pressure suppression.** Containment pressure suppression (and radionuclide scrubbing) can be accomplished by either of two systems: the containment spray system or the safety grade containment fan coolers. The two containment spray pumps are used in injection; for recirculation the spray headers get their water from the RHR or LPR pumps (see above). In case of failure of RHR heat exchangers, the containment fan coolers provide a redundant means of containment heat removal. The containment fan coolers can accomplish

their mission in an accident if three out of five coolers work during the injection and/or recirculation phase.

The following features would tend to increase the CDF:

- **Requirement for active ventilation for auxiliary feedwater pump cooling.** An active ventilation system is used to cool these pumps. Loss of this ventilation system requires the operators to manually open a roll-up door to preserve adequate room cooling.

- **Requirement for cooling of the LPR pumps.** These pumps must be cooled by the auxiliary component cooling system (injection phase of an accident) or the component cooling system (recirculation phase).

- **Manual switchover from injection to recirculation phase.** Operator action is required to switch from injection mode to recirculation mode.

- **Block valves are closed a certain fraction of time.** Technical specifications do not prevent startup with the block valves closed. Operating personnel have been apprised of the enhanced risk importance of these components.

- **Emergency diesel generator cooling.** The EDG require cooling by service water.

- **Emergency diesel generator ventilation.** This is a required function, however the EDG building ventilation can receive power only from EDG 21 and EDG 22. Therefore startup of only EDG 23 would not result in a success of the emergency power system due to lack of the necessary ventilation for continued EDG 23 operation. The licensee has recently changed the configuration to allow the connection of EDG 23 to the ventilation system.

Indian Point 2 has a large, dry containment constructed of reinforced concrete with a steel liner. Indian Point 2 is comparable to Zion in power level, containment free volume, reactor coolant system (RCS) water volume, fuel clad mass, and containment failure pressure. According to the submittal, the Indian Point 2 containment and the Zion containment are very similar in interior layout and equipment location, and have often been treated as being essentially the same in past analyses.

The following plant-specific features are important for accident progression in the Indian Point 2 plant:

- A cavity geometry which promotes the flow of containment water to the reactor cavity as well as transportation and dispersion of core debris into the lower compartment. The cavity is connected to the lower compartment by an open inclined "chute" through which the instrumentation leads pass, an open vertical access shaft located immediately adjacent to the biological shield, and the annulus between the reactor vessel and the biological shield which

leads to the penetrations for the hot and cold legs. The cavity has a large area for corium spreading, comparable to that of Zion.

- A significant degree of diversity and redundancy in the containment heat removal systems. Sufficient containment heat removal can be provided by the containment spray system driven by either a recirculation pump, located in the lower compartment of the containment and taking suction from the recirculation sump, or an RHR pump, located outside the containment and taking suction from the containment sump. Sufficient heat removal can also be provided by three of the five fan coolers, located in the annular compartment below the operating decks. The geometrical factors were considered in assigning the probabilities for loss of function due to harsh environmental conditions in the IPE.

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

- The basaltic nature of the concrete used. Although it is referred to in the IPE submittal as being of a "limestone" type, the concrete is in fact somewhere between the basaltic and medium carbonate types (according to the licensee's response to the RAI).

E.2 Licensee's IPE Process

The licensee initiated work on a probabilistic risk assessment (PRA) for Indian Point 2 in response to Generic Letter 88-20. The freeze data for the analysis is mid-1991, at the end of the refueling outage.

The IPE was performed by Con Edison staff with support from Halliburton NUS Corporation. ConEd staff managed the IPE and participated in the analysis. HNUS, managed the technical part of the project, and provided leadership in some technical areas.

To support the IPE process, a review was made of, and models were built upon, a previous PRA of Indian Point 2, the Indian Point Probabilistic Safety Study (IPPSS). Attention was also focused on other PRAs for plants similar to Indian Point 2.

The extent of utility personnel involvement in the HRA was not discussed. Procedure reviews, discussions with operations and training staff, and observations of simulator training sessions helped assure that the IPE HRA represented the as-built, as-operated plant. An in-house technical review of the IPE was performed by Con Edison personnel not involved in the PRA. Samples of reviewer comments provided in the submittal indicated the HRA was reviewed from a technical point of view. 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 and several potential human performance related improvements were identified and discussed.

The analysis was reviewed at several levels. A formal, independent review (by plant personnel not involved in the PRA) was also performed. The IPE team responded to and resolved comments from this formal review.

Samples of reviewer comments provided in the submittal indicated the HRA was reviewed from a technical point of view. 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 and several potential human performance related improvements were identified and discussed.

The submittal does not explicitly indicate whether the licensee intends to maintain a "living" PRA, although it is implied that the IPE will be periodically updated.

E.3 IPE Analysis

E.3.1 Front-End Analysis

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

The IPE quantified the following initiating event categories: three LOCAs, 7 general transients and 12 special initiators. The IPE developed 8 event trees for frontline systems/functions and two support system event trees to model the plant response to these initiating events. No flooding analysis was performed.

Success criteria were based on WCAP analyses and runs with MAAP codes analyzing thermal-hydraulic plant models.

Like some other PWR IPEs, the Indian Point 2 IPE assumes that if high head injection fails following a small LOCA, the primary system can be depressurized via the secondary system sufficiently fast so that the accident can be mitigated with low head safety injection and the accumulators. This assumption reduces the CDF from a small LOCA.

The RCP seal cooling model assumes that both CCW and seal injection must fail in order for the seals to fail. This element of the success criteria is consistent with other PWR PRA studies.

The data collection process period was through mid-1991. This means that the data used in the original IPPSS of 1982, its amendments of 1982 and 1983 and the PRA update of 1989 were updated with data for the period 1989-1991. Plant specific component failure data were used to update generic data with the use of Bayesian techniques. Plant specific data were used exclusively for unavailabilities due to test and maintenance activities.

IP-2 data are generally in agreement with the NUREG/CR-4550 data, except for the emergency diesel generator failure to start, which is one order of magnitude below the NUREG value.

The multiple Greek letter (MGL) approach was used to characterize common cause failures. Some MGL parameters seem low (HHSI pumps, MOVs and EDGs), however they are based on extensive plant specific considerations.

The internal core damage frequency has a mean of 3.13E-5/yr, and 5th, 50th and 95th percentiles of 1.63E-5, 2.76E-5 and 5.28E-5/yr, respectively. This CDF distribution is unusually tight (error factor less than 2) and therefore the distribution appears questionable. However, a Level 1 uncertainty analysis was not requested in Generic Letter 88-20 or NUREG-1335. The internal accident types and initiating events that contribute most to the CDF and their percent contributions are listed below in Tables E-1 and E-2:

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

Initiating Event Group	Contribution to CDF (/yr)	%
General Transients	1.2E-5	37.3
LOCAs	1.0E-5	33.3
Station Blackout	4.5E-6	14.3
Anticipated Transients Without Scram	1.8E-6	5.8
Loss of Support System Initiating Event	1.3E-6	4.1
Steam Generator Tube Rupture without Stuck Open Relief Valve	1.2E-6	4.0
Steam Generator Tube Rupture with Stuck Open Relief Valve	3.7E-7	1.2
Interfacing System LOCA	2.7E-8	0.09
TOTAL CDF	3.1E-5	100.0

Table E-2. Initiating Events and Their Contribution to the CDF[1]

Initiating Event	Contribution to CDF (/yr)	%
Loss of Offsite Power	5.9E-6	19.0
Small LOCA	5.7E-6	18.1
Reactor Trip	4.4E-6	14.0
Loss of Feedwater	2.7E-6	8.7
Turbine Trip	2.7E-6	8.6
Large LOCA	2.6E-6	8.2
Medium LOCA	1.9E-6	6.1

The IPE assigned a Plant Damage State (PDS) to the end of each Level 1 event sequence as the interface between Level 1 and Level 2 portions of the analysis.

E.3.2 Human Reliability Analysis

The HRA process for the Indian Point 2 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 submittal indicated that the analysis of pre-initiator actions included both miscalibrations and restoration faults. A numerical screening analysis was not conducted on pre-initiator human actions, but it appeared that at least in one case, a qualitative screening of a miscalibration event did occur. All identified pre-initiator actions purportedly received detailed quantification using the NRC Human Reliability Handbook [NUREG/CR-1278] and appropriate dependencies were considered. However, documentation of the quantification of miscalibration related events was limited. No pre-initiator human actions were identified among the important human actions.

The Indian Point 2 IPE did not make an explicit distinction between response and recovery type post-initiator human actions. However, the submittal states that no credit was taken for actions that were not supported by written operating procedures and that have not been demonstrated to be viable. Given this assertion, the submittal addresses and quantifies all post-initiator human actions with the same approach and methodology. Quantification of post-initiator human actions was based primarily on the EPRI *Generalized Event Tree Representation of Post-Accident Actions* [EPRI NP-6937], in conjunction with EPRIs operator reliability experiments (ORE) based methodology [EPRI NP-6560-L]. Three parameters are used in the quantification process: the probability of unrecovered detection errors (p_1), the probability of non-response within available time (p_2), and the probability of failure

[1]Only the most dominant initiating event contributors to the CDF are listed here.

to correctly execute the appropriate step in a procedure (p_3). The method selects some base probabilities (e.g., for the response execution parameter (p_3)) from the NRC Handbook [NUREG/CR-1278]. In addition, in the licensee's response to the NRC RAI, it was stated that for cases in which "the crew were not on a time critical path," a nominal value for probability of diagnosis (p_2) of 2.7E-3 was applied based on the ASEP HRA methodology [NUREG/CR-4772].

Plant-specific performance shaping factors and dependencies (such as those among multiple actions in a sequence) were considered. Human errors were identified as important contributors in accident sequences leading to core damage and several potential human performance related enhancements were identified. No major limitations with the post-initiator analysis were identified.

E.3.3 Back-End Analysis

Plant Damage States (PDSs) are used as the initial conditions for the Level 2 analysis. The PDSs are defined by the use of a logic diagram (an event tree) with headings that describe the RCS and containment system status.

Quantification of accident progression in the Indian Point 2 IPE involves the development of a small containment event tree (CET), with 8 top events, and the development of the decomposition event trees (DETs) for each of the eight CET top events. The CET and its supporting DETs developed in the IPE addressed all the containment failure modes discussed in NUREG-1335. However, there is a slight inconsistency in the Indian Point 2 CET/DET logic structure. While the effect of RCS depressurization due to hot leg creep rupture is considered in the CET, its effect on early containment failure is not included in the early failure DET. This may have either a positive or a negative effect on early containment failure probability. For the early failure mechanisms considered in the IPE, the probability of failure due to in-vessel steam explosion (the Alpha mode failure) increases with RCS depressurization and that due to the phenomena associated with high pressure melt ejection (HPME) decreases with RCS depressurization. Since the contributions from both of these early failure mechanisms are predicted to be small in the IPE, the omission of RCS depressurization due to hot leg failure is not expected to have a significant effect on the containment failure profile for Indian Point 2.

Although the Indian Point 2 CET structure addresses all the containment phenomena and containment system status information important to containment performance, it does not include any recovery actions. AC power recovery is considered in the IPE as a PDS parameter and is credited only for SBO sequences. The recovery actions that are considered in some other IPEs, such as RCS depressurization by operator actions and recovery of containment heat removal capability, are not modeled in the CET structure of the IPE. This lack of consideration of recovery actions in the CET can be construed as conservative, but it limits the use of the CET to investigate the effectiveness of operator recovery actions on containment performance, which may be of interest for the development of accident management strategies.

The quantification of the Indian Point 2 CET and its supporting DETs relies heavily on the results found in the NUREG-1150 studies. This is supplemented by the data obtained from plant-specific MAAP analyses, and, in some cases, by the engineering judgment of the analyst. In general, the quantification process for the CET and the associated DETs is systematic and traceable.

The results of the CET analyses lead to an extensive number of CET end states, which are binned into 26 source term categories (STCs). The 26 STCs are further grouped into five release category types. The grouping of release category types is based on the release of iodine and cesium release fractions to facilitate comparison with GL 88-20 sequence selection criteria in terms of release magnitudes. Among the five release types, Types I and II have release fractions which exceed those defined in WASH 1400 for the PWR-4 release category. The CDF contribution from these two release types is about 10%. Release fractions for the STCs are determined from the analyses of representative sequences using the MAAP computer code.

For the Indian Point 2 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.

The leading PDS involves a general transient with the RCS at system pressure, with no in-vessel injection and with both containment spray and containment heat removal available (39% of CDF). This is followed by a PDS with a LOCA initiator, with the RCS in a pressure range from 200 to 2,000 psig, without in-vessel injection and containment spray, but with containment heat removal (18% of CDF). For Indian Point 2, the PDSs with RCS pressure at system pressure level contribute to about 50% of CDF, and the PDSs with low RCS pressure (i.e., less than 200 psig) contribute less than 10% of CDF.

Table E-3 shows the probabilities of containment failure modes for Indian Point 2 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	Indian Point 2	Surry	Zion
Early Failure	0.13	0.7	1.4
Late Failure	9.0	5.9	24.0
Bypass	6.2	12.2	0.7
Isolation Failure	0.05	*	**
Intact	84.6	81.2	73.0
CDF (1/ry)	3.1E-5	4.0E-5	3.4E-4

* Included in Early Failure, approximately 0.02%
** Included in Early Failure, approximately 0.5%

Of the 0.0013 conditional probability of early containment failure, about half comes from SBO sequences and another half from the other transient initiated sequences. The contribution to early containment failure from LOCA sequences is negligible.

Containment bypass failure comes primarily from SGTR initiated events (86% of bypass CDF). This is followed by induced SGTR (13% of bypass CDF) and ISLOCA (1% of bypass CDF). Of these three bypass failures, only induced SGTR is affected by Level 2 accident progression. For induced SGTR, NUREG-1150 data are used in the Indian Point 2 IPE to determine failure probability. The effect of restarting the RCPs, which is considered in some other IPEs as a mechanism that may increase the potential of induced SGTR, is not addressed in the Indian Point 2 IPE.

Of the 0.09 conditional probability for late containment failure, 0.035 is from the LOCA sequences, 0.03 from the transient sequences, and 0.025 from the SBO sequences. On a conditional basis, it is more likely for an SBO sequence to develop a late failure (about 16% of SBO CDF) than LOCA sequences (about 12% of LOCA CDF) and other transient sequences (about 6% of transient CDF).

Release fractions for the Indian Point 2 IPE are obtained from MAAP calculations. MAAP calculations were performed in the IPE for eight sequences, and release fractions for the 26 source term categories are characterized by their similarity to one of the eight MAAP calculations. The sequences selected for MAAP calculations include an ISLOCA sequence, two SGTR sequences with and without a stuck-open valve, two SBO sequences with early containment rupture or leak, and three small LOCA sequences with late containment failure and with various core injection and containment spray availability status. Detailed discussions of how these sequences are selected to represent the source term categories are not provided in the submittal. Nonetheless, the selected sequences seem to provide a reasonable representation of the source term categories. Furthermore, in the IPE submittal, the release fractions obtained from the MAAP calculations are compared with those obtained from previous studies (e.g., NUREG-1150 for Surry). The comparison shows reasonable agreement.

The sensitivity study provided in the Indian Point 2 IPE seems to have addressed the issues of significant uncertainty in the analysis. The sensitivity analysis shows that early containment failure is not sensitive to the assumptions related to DCH load and containment pressure capability. This is a result of the large volume and high pressure capability of the Indian Point 2 containment which lead to a small probability of early containment overpressure failure. However, late containment overpressure failure is sensitive to containment spray availability, and late basemat melt-through is sensitive to the assumption of ex-vessel debris coolability. The latter is not a significant concern because of the significant uncertainties associated with debris coolability and the long time required for basemat melt-through. Late overpressure may not be a concern for Indian Point 2 because of the diversity of the containment spray system.

E.4 Generic Issues and Containment Performance Improvements

The IPE addresses decay heat removal (DHR). CDF contributions were estimated for the following DHR methods: auxiliary feedwater system (AFW), primary bleed (also referred to as feed and bleed) and low and high recirculation cooling. Failure of the AFW and primary bleed was determined to make a major contribution to the total CDF. The AFW failures are dominated by hardware failures in the motor driven and turbine driven pumps. A major contributor to primary bleed failure is operator error. The licensee is emphasizing primary bleed in operator training and analyzing the possibility of increasing the time window by providing an earlier cue for feed and bleed in the EOPs. The absolute CDF contribution of the AFW and the primary bleed is similar to Seabrook and Surry and therefore the licensee concludes there is no vulnerability.

No other generic issues were analyzed in this IPE.

The CPI recommendation for PWRs with a dry containment is to evaluate containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements. A level 2 plant walkdown was performed by Indian Point 2 IPE PRA personnel. This was augmented by examination of a video laser disc walkdown record that was available for areas not considered accessible because of high radioactivity or other reasons. No areas were found in the containment where significant pockets of hydrogen could form and the different volumes in the containment were found to be well connected. Local detonations were postulated not to occur due to rapid mixing that would ensue within containment volumes.

The effect of harsh environmental conditions on equipment survivability is assessed in some detail in CET quantification. Although the quantification involves significant engineering judgment, the assessment is logical and reasonable. Additionally, there is a significant degree of diversity and redundancy in the Indian Point 2 containment heat removal systems, and they are geometrically separated inside and outside the containment.

E.5 Vulnerabilities and Plant Improvements

The licensee used the NUMARC severe accident guidance criteria to search for vulnerabilities. No vulnerabilities were found.

The IPE took credit only for plant modifications and improvements that are complete. The following four modifications were made in response to conclusions drawn in the IPE and have been completed (but have not been credited in the analysis):

1. Gas Turbine No. 2 (which is the most reliable of the three gas turbines) has been provided with the full blackstart capability. This will provide an additional means of recovering power following station blackout sequences. The estimated change in the core damage frequency is minimal (no more than a few percent reduction, per RAI Responses).

2. A sixth fan has been installed in the EDG building and the power supply configuration to the EDG building fans has been improved to eliminate the earlier dependency between failures of EDG 21 and 22 (which had been used to power all the fans in the EDG Building) and EDG 23 (which previously had no such capability). Now each EDG powers two of the fans, with one fan per EDG being sufficient to provide ventilation. An estimate of a 5% expected reduction in the CDF due to this modification is provided in the RAI Responses.

3. Periodic testing of all EDG Building fans has been instituted.

4. The operating position of the PORV Block Valves is being tracked to provide a fuller basis for modeling. Operating and technical personnel have been made aware of the risk importance of these valves. Moreover, PORVs with a more leak resistant design have been installed during the 1995 refueling outage.

In addition, the following EOP modification is under consideration:

5. Changing the EOPs to enable an earlier initiation of feed and bleed is being considered. There are possible downside effects from this action in that this may detract from operator efforts to recover the more benign method of core cooling, i.e. secondary cooling by recovering either the Auxiliary Feedwater or the Main feedwater/condensate system. Therefore it is not clear that this modification will be implemented.

No quantitative impact of these changes on the CDF is available at this time but the licensee intends to incorporate modeling of the above modifications into the next PSA update (RAI responses).

No containment vulnerabilities were identified as a result of performing the IPE and no back-end plant improvements are mentioned in the IPE submittal.

E.6 Observations

The licensee appears to have analyzed the design and operations of Indian Point 2 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 Indian Point 2; gained a quantitative understanding of the overall frequency of core damage; and implemented changes to the plant to help prevent and mitigate severe accidents.

Strengths of the level 1 IPE analysis are as follows: Plant specific data were used where possible to support the quantification of initiating events and component unavailabilities.

No major weaknesses of the level 1 analysis were identified. Some common cause MGL factors seem low compared to NUREG/CR-4550 data, however the licensee seems to have done a thorough analysis of common cause data as it applies to this plant, along with a cause-defense analysis.

The IPE determined that failures in the Auxiliary Feedwater system (dominated by hardware pump failures) and in the primary bleed (dominated by operator failures and failure of PORVs to open) are the primary contributors to core damage.

LOCAs are a relatively high contributor to the CDF (33%) due to operator error in aligning recirculation.

The contribution of station blackout (14%) is due to some peculiar plant features. On the one hand, electric power recovery is enhanced by the fact that two of the three gas turbines have blackstart capability (at the time of the IPE). On the other hand, distribution of power from the three EDGs to the safeguards equipment is staggered (i.e. not every EDG will power every system) and the start of EDG 23 only will not constitute success because of the arrangement of the power supply to the EDG Building fans. The number of EDGs is not always matched with the number of trains of a safety system (e.g. EDG 21 will not power a MDAFW pump as there are only two such pumps, connected to the buses powered by the other two EDGs). The resistance of the plant to station blackout has been improved by adding blackstart capability to the third gas turbine and correcting the alignment of the EDG-to-EDG building fan power supply.

As noted above, several improvements have been completed as a result of insights from the IPE. The CDF impact of these improvements is not known.

The HRA review of the Indian Point 2 IPE submittal did not identify any significant problems or errors. A viable approach was used in performing the HRA and although there were some minor potential weaknesses, 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 indicates that utility personnel were involved in the HRA and that the walkdowns, documentation reviews and simulator observations represented a viable process for confirming that the HRA portions of the IPE represent the as-built-as operated plant.

2) The submittal indicated that the analysis of pre-initiator actions included both miscalibrations and restoration faults. Eleven restoration events and their HEPs were listed in the submittal. In response to an NRC RAI, treatment of miscalibration events was discussed in more detail than in the submittal, but exactly how miscalibration HEPs were determined was not clearly explained. In addition, a listing of all of the miscalibration events was not provided and it was unclear exactly how many were actually modeled. Treatment of miscalibration events could be a minor weakness of the pre-initiator analysis.

3) The Indian Point 2 IPE did not make an explicit distinction between response and recovery type post-initiator human actions. However, the submittal states that no credit was taken for human actions that were not supported by written operating procedures and that were not demonstrated to be viable. A screening analysis of the post-initiator events was not conducted. All events

received detailed quantification and dependencies and context effects were considered. The quantification method and its application was acceptable and the overall analysis appeared reasonably thorough. The only potential weaknesses in regards to the post-initiator analysis concerned the extension of the basic methodology to better account for PSFs and dependencies. While the "extensions" seemed to improve the methodology, evidence for the validity of the assumptions underlying the improvements were not provided. Nevertheless, the extensions did have significant face validity, i.e., they appeared reasonable.

4) Manual backups to failed automatic actions were conservatively not modeled. The IPE submittal noted that this facilitates consideration of dependencies, which is true. However, by not taking credit for manual recoveries of failed automatic actions, the results of the IPE could be somewhat distorted. To assume that such recoveries would fail may be conservative, but is not realistic. At a minimum, such an approach prevents the licensee from getting the full benefits of the IPE and may impact the discovery of issues related to accident management.

5) The licensee did not identify important human actions through the use of importance measures in the submittal. The submittal did provide a sufficient discussion of operator actions in dominant sequences and a sensitivity analysis for human action events in truncated sequences was performed and discussed. Thus, information regarding important human actions was provided.

The assessment of the level 2 analysis review is that the Indian Point 2 IPE submittal documentation, and the responses to the RAI, contains substantial back-end information regarding the severe accident vulnerability issues for the Indian Point 2 plant.

The following are the major findings of the level 2 analysis described in the submittal:

- The containment analyses indicate that there is a 15% conditional probability of containment failure. The conditional probability of containment bypass is 6.2%, and the conditional probability of early containment failure is 0.13%.

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

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

- The licensee has addressed the recommendations of the CPI program.

The strengths of the back-end analysis include the following:

- The CET and its supporting DETs are well structured and easy to understand. The use of the NUREG-1150 data for CET quantification seems justifiable. In places where engineering

judgment is used, the approach seems logical and the assigned values seem reasonable. Furthermore, parameters with significant uncertainties are included in the sensitivity study.

The weaknesses of the back-end analysis include the following:

- Accident sequences are selected in the IPE for MAAP calculations to provide data to assist CET quantification and for estimating the source terms. However, the selection criteria is not discussed in the IPE submittal. The relationship between the selected sequences and the accident sequences binned to the PDSs or the source term categories is not established or discussed in the submittal. Nonetheless, the sequences selected for MAAP calculation seem to provide a reasonable representation of the source term categories.

- While the effect of RCS depressurization due to hot leg creep rupture is considered in the CET, its effect on early containment failure is not included in the early failure DET. Although the contributions from the mechanisms that are affected by RCS depressurization are not significant in the Indian Point 2 IPE, (i.e. this omission is not expected to have a significant effect on the containment failure profile), it does represent an inconsistency in the CET modeling.

- Except for the recovery of ac power for the SBO sequences, operator recovery actions, such as those for RCS depressurization and recovery of containment heat removal systems, are not included in the IPE. Although this may provide a conservative estimate of accident progression results, it limits the use of the CET structure to investigate the likely benefit of these recovery actions.

NOMENCLATURE

AFW	Auxiliary Feed Water
CCF	Common Cause Failure
CCI	Core-Concrete Interaction
CCW	Component Cooling Water
CDB	Core Damage Bins
CDF	Core Damage Frequency
CET	Containment Event Tree
ConEd	Consolidated Edison
CPI	Containment Performance Improvement
CS	Containment Spray
CST	Condensate Storage Tank
DCH	Direct Containment Heating
DET	Decomposition Event Tree
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
FTC	Failure to Close
FTO	Failure to Open
FTR	Failure to Run
FTS	Failure to Start
GL	Generic Letter
GSI	Generic Safety Issue
HEP	Human Error Probability
HHSI	High Head Safety Injection
HPI	High Pressure Injection
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
IPPSS	Indian Point Probabilistic Safety Study
ISGTR	Induced Steam Generator Tube Rupture
ISLOCA	Interfacing System LOCA
IPE	Individual Plant Examination
LER	Licensee Event Report
LPI	Low Pressure Injection
LPR	Low Pressure Recirculation
MDAFW	Motor Driven AFW
MOV	Motor Operated Valve
MGL	Multiple Greek Letter
Mwt	Megawatt thermal

ORE	Operator Reliability Experiments
PDS	Plant Damage State
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RC	Release Category
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SFAS	Safety Features Actuation System
SIS	Safety Injection System
SGTR	Steam Generator Tube Rupture
SW	Service Water
TDAFW	Turbine Driven AFW
TER	Technical Evaluation Report
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 Indian Point 2 Individual Plant Examination (IPE) submittal [IPE, RAI Responses]. This technical evaluation report adopts the NRC review objectives, which include the following:

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

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

A Request of Additional Information (RAI), which resulted from a preliminary review of the IPE submittal, was prepared by BNL and discussed with the NRC on February 8,1995. Based on this discussion, the NRC staff submitted an RAI to the Consolidated Edison Company of New York on April 26, 1995. Consolidated Edison Company of New York responded to the RAI in a document dated October 31, 1995. This TER is based on the original submittal and the response to the RAI (RAI Responses).

The submittal states that, with NRC concurrence, the risk from internal flooding is not analyzed in this IPE and will be dealt with later, as part of the IPEEE process which considers the risks from external events (fire, flood, seismic events, etc.).

1.2 Plant Characterization

The Indian Point 2 Nuclear Power Plant is a Westinghouse 4 loop pressurized water reactor (PWR). The plant has a power rating of 3083 Mwt (the plant's rating has been increased by 11.5% from the original value). IP-2 is operated by Consolidated Edison Company of New York. The plant went into commercial operation in 1974.

Collocated on the site is Indian Point Unit 3 (3,025 Mwt), operated by the New York Power Authority, and the older, decommissioned Unit 1. No mention is made in the submittal of any dual unit effects between Units 2 and 3 and responses to the RAI indicate that Indian Point 2 and Indian Point 3 are basically independent of each other. In an emergency some auxiliary systems from Unit 1 can be used. The Indian Point site is located on the east bank of the Hudson River, in Westchester County, New York.

A number of design features at Indian Point 2 impact the core damage frequency. However, the submittal does not highlight all of these features nor does it estimate their effect on the core damage frequency. The following features tend to decrease the CDF:

- **Feed and bleed capability.** This capability is not singled out in the IPE as being important. However, it is credited in the event trees and is proceduralized, according to the IPE. However, the time available is somewhat limited, and the plant is considering changing the EOPs to require feed and bleed initiation at an earlier point in time.

- **City water supply connection to the AFW system.** The 1.5 million gallon City Water Storage Tank can be aligned to the AFW pumps should the 360,000 gallon condensate storage tank (CST) become unavailable or exhausted. This is not mentioned in the IPE as being an important design feature.

- **Manual AFW control following station blackout.** The design of the turbine driven AFW pump (TDAFW) permits the operator to manually control it even when DC power has failed (in a prolonged blackout). Pneumatic instrumentation, independent of ac and dc power sources, is available for the pressurizer and the steam generator, so that stable plant conditions can be maintained in the event of a prolonged station blackout even after battery capacity has been exhausted. Ventilation and cooling to the area housing the AFW equipment can be provided by opening the roll up door which gives direct access to the transformer yard. The IPE mentions the manual control of AFW as an important design feature.

- **City water/primary water cooling of HPI, RHR and charging pumps.** The normal source of cooling for the RHR, SIS (i.e. HPI) and charging pumps is the component cooling water system. However, in the event of a loss of component cooling, a hard-piped city water pump cooling connection to the charging pumps is available, and the procedure specifically directs the operator to make the proper alignment. At the same time, the procedure directs the operator to align the HPI and the RHR pump cooling to either the city water or the hard-piped primary makeup water connection. In addition, the RHR (i.e. the LPI) and the HPI pumps may be operated in the injection mode without cooling, whereas the charging pumps can be operated at maximum speed to extend the time they may be operated without cooling.

  Thus the potential for an RCP seal LOCA, or a LOCA associated with a loss of CCW or non-essential service water, is significantly diminished.

- **Emergency power system.** IP-2 has three emergency diesel generators plus three gas turbines, which can be used if the diesel generators fail. Gas turbines 1 and 3 have full black start capability. Gas turbine 1 is located on the IP-2 site, whereas gas turbines 2 and 3 are located close by at the Buchanan substation. The operating procedures direct the operators to start up and connect the gas turbines in case of an accident. The plant will be adding the blackstart capability to gas turbine 2.

- **Recirculation capability.** Two sets of low pressure pumps are provided for recirculation: two RHR pumps(outside the containment) and two LPR pumps (inside the containment).

The water is recirculated from the sump, through the RHR heat exchangers and into the RCS, and/or the spray headers for containment pressure control.

- **Service water.** The essential and the non-essential service water headers may be aligned to different service water pumps in case of pump failure (each SW system has 3 pumps). This is not mentioned in the IPE as an important feature.

- **High to Low pressure piping interface.** Since the high to low pressure interface on the RHR system is inside the containment, there is a strong possibility that, should an interfacing system LOCA due to failure of suction valves, it will occur inside the containment. This significantly reduces the frequency of containment bypass events.

- **Containment pressure suppression.** Containment pressure suppression (and radionuclide scrubbing) can be accomplished by either of two systems: containment spray system or the safety grade containment fan coolers. The two containment spray pumps are used in injection; for recirculation the spray headers get their water from the RHR or LPR pumps (see above). In case of failure of RHR heat exchangers, the containment fan coolers provide a redundant means of containment heat removal. Containment fan coolers can accomplish their mission in an accident if three out of five coolers work during injection and/or recirculation phase.

The following features would tend to increase the CDF:

- **Requirement for active ventilation for auxiliary feedwater pump cooling.** An active ventilation system is used to cool these pumps. Loss of this ventilation system requires the operators to manually open a roll-up door to preserve adequate room cooling.

- **Requirement for cooling of the LPR pumps.** These pumps must be cooled by the auxiliary component cooling system (injection phase of an accident) or the component cooling system (recirculation phase).

- **Manual switchover from injection to recirculation phase.** Operator action is required to switch from injection mode to recirculation mode.

- **Block valves are closed a certain fraction of time** (about 25% of the time for each valve). Technical specifications do not prevent startup with the block valves closed. Operating personnel have been apprised of the enhanced risk importance of these components.

- **Emergency diesel generator cooling.** The EDGs require cooling by service water.

- **Emergency diesel generator ventilation.** This is a required function, however the EDG building ventilation can receive power only from EDG 21 and EDG 22. Therefore startup of only EDG 23 would not result in a successful operation of the emergency power system due to a lack of necessary ventilation for the continued operation of EDG 23. The

licensee has recently changed the configuration to allow the connection of EDG 23 to the ventilation system.

Indian Point 2 has a large, dry containment constructed of reinforced concrete with a steel liner. 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 Indian Point 2 Nuclear Station

Characteristic	Indian Point 2	Zion	Surry
Thermal Power, MW(t)	3071	3236	2441
RCS Water Volume, ft^3	12,250	12,700	9200
Containment Free volume, ft^3	2,610,000	2,860,000	1,800,000
Mass of Fuel, lbm	222,000	216,000	175,000
Mass of Zircalloy, lbm	41,994	44,500	36,200
Containment Design Pressure, psig	47	60	45
Median Containment Failure Pressure, psig	126	135	126
RCS Water Volume/Power	4.0	3.9	3.8
Containment Volume/Power	850	884	737
Zr Mass/Containment Volume, lbm/ ft^3	0.016	0.016	0.020
Fuel Mass/Containment Volume, lbm/ ft^3	0.085	0.076	0.097

Indian Point 2 is comparable to Zion in power level, containment free volume, reactor coolant system (RCS) water volume, fuel clad mass, and containment failure pressure. According to the submittal, the Indian Point 2 containment and the Zion containment are very similar in interior layout and equipment location, and have often been treated as being essentially the same in past analyses.

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

- A cavity geometry which promotes the flow of containment water to the reactor cavity as well as transportation and dispersion of core debris into the lower compartment. The cavity is connected to the lower compartment by an open inclined "chute" through which the instrumentation leads pass, an open vertical access shaft located immediately adjacent to the biological shield, and the annulus between the reactor vessel and the biological shield which

leads to the penetrations for the hot and cold legs. The cavity has a large area for corium spreading, comparable to that of Zion (about 1,000 square feet).

- A significant degree of diversity and redundancy in the containment heat removal systems. Sufficient containment heat removal can be provided by the containment spray system driven by either a recirculation pump, located in the lower compartment of the containment and taking suction from the recirculation sump, or an RHR pump, located outside the containment and taking suction from the containment sump. Sufficient heat removal can also be provided by three of the five fan coolers, located in the annular compartment below the operating decks. The geometrical factors, like relative location, were considered in assigning the probabilities to loss of function of these systems due to harsh environmental conditions.

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

- The basaltic nature of the concrete used. Although it is referred to in the IPE submittal as being of a "limestone" type, the concrete is in fact somewhere between the basaltic and medium carbonate type (according to the licensee's response to Question 30 of the RAI).

2. TECHNICAL REVIEW

2.1 Licensee's IPE Process

The process used by the licensee was reviewed with respect to: completeness and methodology; multi-unit effects and as-built, as-operated status; and licensee participation and peer review.

2.1.1 Completeness and Methodology

The 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 large event tree/small fault tree support state technique, and it was clearly described in the submittal.

Event trees were developed for all classes of initiating events considered. A limited top event importance analysis has been performed and is described in the submittal. In addition, a sensitivity analysis was performed to assess the change in the core damage frequency point estimate if the original IPPSS RCP seal LOCA model was used. Another sensitivity analysis was performed to assess the impact of a more realistic turbine driven auxiliary feedwater pump mission time. An uncertainty analysis was performed that provided a probability distribution for the total CDF. A third sensitivity analysis raised all human error probabilities (HEPs) to 0.1 to find any potentially important accident sequences screened out due to the HEPs used.

To support the IPE process, the licensee made a review of, and built the model upon, a previous probabilistic study on Indian Point 2, the Indian Point Probabilistic Safety Study (IPPSS). Consideration was also given to NRC review comments on this study. Other PRA studies were also reviewed. Examples are the Surry IPE and the Surry NUREG-1150 study, the Seabrook PRA and IPE, the Zion NUREG-1150 study and the ZPSS, and the Turkey Point 3,4 IPE.

The submittal information on the HRA process was generally complete in scope. Some additional information/clarification was obtained from the licensee through an NRC request for additional information. The HRA process for the Indian Point 2 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). Eleven restoration events and their HEPs were listed in the submittal. In response to an NRC RAI, treatment of miscalibration events was discussed in more detail than in the submittal, but exactly how miscalibration HEPs were determined was not clearly explained. In addition, a listing of all of the miscalibration events was not provided and it was unclear exactly how many were actually modeled. Treatment of miscalibration events could be a minor weakness of the pre-initiator analysis.

The Indian Point 2 IPE did not make an explicit distinction between response and recovery type post-initiator human actions. However, the submittal states that no credit was taken for actions that were not supported by written operating procedures and that have not been demonstrated to be viable. Given this assertion, the submittal addresses and quantifies all post-initiator human actions with the same approach and methodology. Quantification of post-initiator human actions was based primarily on the EPRI *Generalized Event Tree Representation of Post-Accident Actions* [EPRI NP-6937], in conjunction with EPRIs operator reliability experiments (ORE) based methodology [EPRI NP-6560-L]. Three parameters are used in the quantification process: the probability of unrecovered detection errors (p_1), the probability of non-response within available time (p_2), and the probability of failure to correctly execute the appropriate step in a procedure (p_3). The method selects some base probabilities (e.g., for the response execution parameter (p_3)) from the NRC Handbook [NUREG/CR-1278]. In addition, in the licensee's response to the NRC RAI, it was stated that for cases in which "the crew were not on a time critical path," a nominal value for probability of diagnosis (p_2) of 2.7E-3 was applied based on the ASEP HRA methodology [NUREG/CR-4772].

Plant-specific performance shaping factors and dependencies (such as those among multiple actions in a sequence) were considered. Human errors were identified as important contributors in accident sequences leading to core damage and several potential human performance related enhancements were identified. No major limitations with the post-initiator analysis were identified.

The Indian Point 2 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 Indian Point 2 IPE submittal for the back-end evaluation is clearly described. Plant Damage States (PDSs) are used as the initial conditions for the Level 2 analysis. The PDSs are defined in the Indian Point 2 IPE by the use of a logic diagram (an event tree) consisting of a set of characteristics which are important to accident progression. Quantification of the accident progression involves the development of a small top level containment event tree (CET), with 8 top events, and decomposition event trees (DETs) for the determination of the CET top events. The CET and its supporting DETs addressed all the containment failure modes discussed in NUREG-1335. The results of the CET analyses lead to an extensive number of CET end states which are binned into twenty six source term categories, which are further grouped into five release categories. The CET quantification relies primarily on NUREG-1150 data. Calculation results from the MAAP 3.0B code analyses were used to provide data to assist in quantifying the CET and for estimating the source terms.

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

There is some sharing of systems between the Indian Point 2 and the Indian Point 3 units (as well as one backup system with the retired Indian Point 1 unit). The shared systems between the two active units are the city water storage tank and the gas turbine (GT-2) located on the IP-2 site. IP-2 has priority access to both of these systems, but both systems are capable of serving both units simultaneously. Therefore, no multi-unit effects are modeled. There is also apparently a possibility

of sharing some ac switchyard equipment, but this is not explicitly described in the IPE submittal or in the RAI responses (IP-3 TER).

The IP-1 river water system is used for cooling the turbine lube oil and other BOP loads, thus removing some of the cooling load from the service water system.

A wide variety of up-to-date information sources were used to develop the IPE, for example the Updated Final Safety Analysis Report (UFSAR), Technical Specifications, WCAP reports, MAAP runs, piping and instrument diagrams, control room and operator log sheets, operating procedures, etc. The freeze date of the analysis was mid-1991 ("plant configuration as existed at the start of the current operating cycle in mid-1991").

Plant specific data were used where possible. In addition, plant walkdowns were performed to clarify information and to identify spatial interactions and dependencies. A plant walkdown was performed to review the HVAC systems in various plant areas. Another walkdown was used in the course of the common cause analysis. A two-day containment walkdown was performed for the Level-2 analysis.

According to the submittal, the IPE builds upon the previous Indian Point Probabilistic Safety Study (IPPSS), which was published in 1982 and amended in 1983, and accounts for additional plant changes and equipment information up through and including the 1991 refueling outage.

Procedure reviews, discussions with operations and training staff, and observations of simulator training sessions helped assure that the IPE HRA represented the as-built, as-operated plant. Since the other Indian Point units are operated by different licensees, little overlap was credited. Credit was not taken for cross-tieing of EDGs, but apparently the potential use of three common gas turbines was credited. The submittal stated that all of the turbines could be controlled from the Indian Point 2 control room.

Insofar as the back-end analyses are concerned, it appears that all the Indian Point 2 containment specific features are modeled.

The submittal does not explicitly indicate whether the licensee intends to maintain a "living" PRA. However, mention is made of the intention to incorporate modeling of the completed plant modifications in the "next PSA update" (RAI Responses).

2.1.3 Licensee Participation and Peer Review

Licensee personnel were involved in the analysis. Consolidated Edison staff managed the IPE, and performed parts of the analysis in conjunction with the contractor, HNUS corporation. The ConEd IPE members were located on site and this facilitated their access to information and equipment needed for the IPE process. The utility IPE team had two permanent members and several additional members who participated in specific tasks. In addition, staff members who had participated in the original IPPSS study were also available for consultation and review. The licensee expressed an

interest in developing and maintaining an in-house PRA capability in order to use the IPE results and insights in future projects.

The IPE was subjected to a dual review process. Approved quality assurance procedures were used which provided for formal review of all work products. This review was performed by both ConEd and HNUS personnel. In addition, independent review was included in the project scope.

The ConEd personnel who provided independent review (i.e. were not part of the original IPE team) had expertise in operations, engineering, safety analysis and risk analysis. There were five utility reviewers, two of which had been involved in the original IPPSS study.

In addition, the analyses for the back-end part of the study were independently reviewed by an outside consultant (Gabor&Kenton).

The independent review seems to have covered most major areas in modeling including HRA, and it seems to have been an ongoing process, i.e. in parallel with the IPE work. Sample reviewers' comments and the IPE team's responses are provided in the report.

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

2.2 Front End Technical Review

2.2.1 Accident Sequence Delineation and System Analysis

2.2.1.1 Initiating Events

The IPE initiating event analysis used the work performed in the IPPSS as a basis, but employed updated data. A few initiating events were added to the ones modeled in the IPPSS, as a result of further consideration of support systems failures. These additional initiating events were the loss of dc power buses 21 and 22 and loss of 6.9 kV buses 2 and 3.

The initiating events analyzed in the IPE are the following:

LOCAs:

- Large LOCA (greater than 6 inches in diameter)
- Medium LOCA (2 to 6 inches in diameter)
- Small LOCA (less than two inches in diameter)

Transients:

- Loss of Main Feedwater (includes loss of instrument air)
- Closure of a Main Steam Isolation Valve
- Loss of Reactor Coolant Flow
- Core Power Excursion

Turbine Trip
Reactor Trip
Loss of Offsite Power

Special Initiators:

Steam Generator Tube Rupture
Steam Line Break Inside Containment
Steam Line Break Outside Containment
Loss of Service Water
Loss of Component Cooling Water
Loss of DC bus D21
Loss of DC bus D22
Loss of 6.9 kV bus 2
Loss of 6.9 kV bus 3
Anticipated Transient Without Scram
Reactor Vessel Rupture
Interfacing Systems LOCA

The support systems whose failure was eliminated from the consideration of initiating events are: station auxiliary transformers, 6.9 kV buses 5 or 6, any 480 V AC buses, any 118 V AC instrument buses, DC buses 23 or 24, the IP-1 River Water System, the HVAC systems, and the City Water System. Loss of any of these systems would not result in an automatic plant trip.

The station auxiliary transformer feeds the 6.9 kV buses 5 and 6. During power operation the other 6.9 kV buses (1, 2, 3 and 4) are supplied from the Unit 2 main generator via the unit auxiliary transformer. Upon the generator trip, a "dead fast transfer" is accomplished such that buses 1, 2, 3 and 4 are fed from the station auxiliary transformer via buses 5 and 6. The submittal and the RAI Responses are not clear why a loss of the station auxiliary transformer or the 6.9 kV buses 5 and 6 would not result in at least a controlled shutdown with some degradation in safety systems (since the four 480 V AC safety buses are fed from the 6.9 kV buses 2, 3, 5 and 6, and all the 6.9 kV buses are fed from the station auxiliary transformer and the buses 5 and 6 upon unit shutdown). The decision to enter a controlled shutdown might be made because a loss of the station auxiliary transformer or the 6.9 kV bus 5 or 6 would result in a degradation of the power supply to the ESF systems, should the ESF systems be needed. The technical specifications require operability of the station auxiliary transformer and the 6.9 kV buses 5 and 6 prior to leaving the cold shutdown state. However, losses of these components would be no worse than a loss of offsite power initiator, with a much lower frequency, therefore these initiators were probably subsumed under LOSP.

Loss of any 480 V AC buses would not result in an automatic plant trip; however neither the submittal nor the RAI Responses specify if a controlled shutdown may ensue, as these buses feed the safety systems. The technical specifications require operability of all four 480 V AC buses prior to leaving the cold shutdown state. However, a loss of a 480 V AC bus would be no worse in consequences than a loss of the associated 6.9 kV bus. Losses of the 6.9 kV buses 2 and 3 were considered and found to contribute very little (<0.1%) to the core damage frequency, therefore a loss of the 480 V AC bus would not be significant either.

Loss of any 118 V AC instrument buses would result in the corresponding ESF channel to go to the fail-safe position (with the exception of the containment spray and the main steam isolation on containment high-high pressure, which is inconsequential, because a LOCA would not be initiated as a result of an instrument bus failure). Therefore, if a controlled shutdown were initiated, it would proceed with all the engineered safety features essentially intact, and therefore it would not be an important initiator due to its low frequency.

Loss of DC buses 23 and 24 would not cause an initiator and would not be significant as these buses are only used as a backup to the DC buses 21 and 22. According to RAI responses, a decision may be made to initiate a controlled shutdown. The technical specifications require availability of all four DC divisions before the unit can be brought above the cold shutdown.

The IP-1 River Water System and the City Water System are backup systems, whose loss would not induce a significant initiator.

Loss of HVAC systems in most areas would not induce a transient due to very slow heat up rates, such that the equipment within would remain below the critical temperatures. In the following three areas the licensee did a more detailed examination with respect to HVAC initiating events:

For the 480 V switchgear room, tests have shown that, with the outside temperature greater than 90°F, it would take more than 6 hours to reach the lowest specified design tolerance of 104°F. The high temperature in the room would be alarmed in the room, with the operating procedure instructing the operators to open the room doors. The room is surveilled frequently (with an entry usually recorded at least once an hour). Due to the slow heat up rate, existence of an alarm and operator procedures to open the door, frequent surveillance and the inherent margin between the design tolerance and the actual thermal capability, a loss of HVAC in this room was excluded from consideration as an initiating event.

Another area which was considered for a potential loss of HVAC initiator is the primary auxiliary building (PAB) housing the RHR and the Safety Injection Pumps. Given operation of the pumps in each room, the RHR Pump 21 room would require restoration of ventilation within 8 hours and the Safety Injection Pump room would require ventilation restoration within 17 hours. However, loss of ventilation in these rooms will not result in a plant trip; furthermore, the Emergency Operating Procedures specifically direct the operators to restore ventilation in these rooms following a transient. Therefore, loss of HVAC in PAB was not considered a credible initiating event.

The third area where a loss of HVAC might be a problem is the EDG building. Ventilation failure in this building is immediately alarmed in the control room. Such a failure is also included in the model for loss of offsite power. However, no transient would be induced by such a failure and therefore loss of EDG Building HVAC was not considered as an initiator.

2.2.1.2 Event Trees

The IPE developed 10 event trees to model the plant responses to internal initiating events. The "general transient" event tree is specialized to account for the many initiating events not specifically assigned a separate event tree. The support system event trees are used in this large event tree-small fault tree approach to model the support system states.

Frontline Event Trees:
- Large LOCA
- Medium LOCA
- Small LOCA
- General Transient with Successful Scram

Special Event Trees:
- Steam Generator Tube Rupture
- Loss of Offsite Power
- Anticipated Transient Without Scram
- Reactor Vessel Rupture

Support System Event Trees:
- Electric Power Event Tree (all AC and DC power)
- Other Support Systems Event Tree (ESFAS, SW, CCW)

Note that the Reactor Vessel Rupture Event tree (or as it is called in the IPE, "the Excessive LOCA event tree", since there is a possibility of multiple pipe failures too) is developed only for the purpose of defining plant damage states. This event is assumed to lead directly to core damage, as is the interfacing systems LOCA (which transfers to only one plant damage state).

The event trees are a mixture of systemic and functional top events. The mission time used in the core damage analysis is 24 hours.

The event tree end states are divided into the two possible core conditions: stable or damaged. A stable core status is defined to include hot shutdown or any other condition where heat removal from the core and from the containment could continue for an extended period of time, with no requirement for additional systems to operate.

Two definitions of core damage were used: 2200°F peak cladding temperature (PCT) or, more conservatively, core uncovery. An RAI response states "...accident analysis documentation does not always indicate when this limit (the PCT) has been exceeded. Where existing analyses addressed PCT, such as for core cooling recovery sequences, and could be applied to the IPE, this was done. For most other cases, use of core uncovery was accepted for events involving loss of RCS inventory, as a conservative substitute for the PCT limit (not including the early stages of most LOCAs which by design may involve short term core uncovery without fuel damage, assuming successful reflood)." (RAI Responses).

Success criteria were based on WCAP analyses and MAAP runs analyzing thermal-hydraulic plant models.

Hot leg recirculation (within 24 hours) is required for a large LOCA. For the medium LOCA, the success criteria specify one high head injection pump in conjunction with three accumulators. If the high head pumps are unavailable, an alternative is to accomplish a rapid secondary/primary depressurization utilizing the AFW and the atmospheric steam dumps, followed by successful ECCS injection and heat removal using the accumulators and one of two RHR pumps. Recirculation may be accomplished using the high head or low head recirculation, depending on the RCS pressure when the RWST is depleted.

Containment heat removal utilizing fan coolers or RHR heat exchangers is required for all LOCAs, including small LOCAs. This is more conservative than criteria found in most PWR IPEs. However, both the containment heat removal and core injection cooling in recirculation can be supplied from one pump (either one recirculation pump or one RHR pump operating in recirculation mode). These pumps can be aligned to the containment spray headers and supply enough flow for both functions.

Small LOCAs also require a reactor trip to shut down the nuclear chain reaction.

Like some other PWR IPEs, the Indian Point 2 IPE assumes that if high head injection fails following a small LOCA, the primary system can be depressurized via the secondary system sufficiently fast so that the accident can be mitigated with low head safety injection and the accumulators (3 out of 4 injecting into the intact legs).

For ISLOCAs, no credit is given to isolation (the event leads to core damage in case of containment bypass), but credit is given to the fact that the high/low pressure interface is inside the containment, and therefore a certain fraction of ISLOCA scenarios will convert to a regular LOCA (i.e., the RCS coolant will be spilled inside the containment).

For transients, in cases where primary bleed cooling (or feed and bleed) is utilized, containment heat removal via fan coolers or RHR heat exchangers is necessary. This is also true when the transient results in a consequential small LOCA. A consequential small LOCA results from a stuck open PORV, an RCP seal failure or an induced steam generator tube rupture. An induced steam generator tube rupture occurs during an uncontrolled steam generator blowdown, which could be due to a steam line break inside containment. In all other cases (e.g. steam line break outside containment, or a stuck open safety valve), flow limiting venturis upstream of the MSIVs prevent excessive blow down rates.

The requirement for containment heat removal for some transient scenarios is conservative compared to other PRAs.

Note that the feed and bleed success requires opening of both PORVs.

The RCP seal cooling model assumes that both CCW and seal injection must fail in order for the seals to fail. This element of the success criteria is consistent with other PWR studies.

The IPE RCP seal LOCA model was based on a Westinghouse analysis (WCAP-10541). Using this model is conservative, according to the submittal, as it applies to unqualified elastomer o-rings, and the plant has switched over to qualified o-rings since the study began. The model yields a probability of $<10^{-3}$ for a seal LOCA at ½ hour following loss of all seal cooling and injection, and which reaches 1 at approximately 10 hours. It should be noted that an improvement in the emergency diesel generator building ventilation system (i.e. eliminating the asymmetrical dependence of the fans on the diesel generators), which was completed after the windup of the IPE, will further reduce the RCP seal failure importance (RAI Responses).

The CCW pumps are used for cooling of the RCP thermal barriers, and also for cooling of the charging pumps. Thus, the CCW pumps support both methods of seal cooling. In case of loss of CCW, the operators are instructed to put the charging pumps on high speed, thus delaying the need for external cooling of these pumps, and to manually align the City Water to provide cooling to the charging pumps. It should be noted that in case of loss of offsite power coincident with an SI signal, the EDG load sequencer does not load the CCW pumps, thus preventing their restart.

In transients, recovery of the turbine driven main feedwater pumps is not credited (these pumps would trip after a turbine trip). Nor is employment of the condenser steam dump as a source of secondary heat removal, for modeling simplification purposes.

Overall, the Indian Point 2 success criteria seem to be somewhat more conservative than most PWR PRAs.

2.2.1.3 Systems Analysis

A total of 16 systems/functions are described in Section 3.2 of the Submittal. Included are descriptions of the following systems: electrical power (AC and DC), ECCS, safeguards actuation, service water, component cooling water, and AFW.

Each system description includes a discussion of the system purpose, a general description (of its operation, operator actions, system interfaces, modeling concerns, etc.), technical specifications for the system, top events considered for the system and split fraction definitions, quantification and equations used in the analysis.

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

The AFW system contains three pumps, two motor driven and one turbine driven. Each motor driven pump supplies two of the four steam generators, while the turbine driven pump supplies all four steam generators. Steam from the turbine driven pump is provided by two of the steam generators (SG 22 and 23).

The AFW pumps are started automatically on low-low level in steam generators, loss of offsite power or loss of main feedwater pumps. Level in the steam generators is controlled manually from the control room. All the AFW pumps and regulating valves can also be operated locally from within the aux feedwater building. In case of a prolonged blackout with loss of DC power, the operators can manually control the TDAFW pump utilizing the pneumatic instrumentation available for the steam generators and the pressurizer.

Two temperature control valves are mounted in series in the steam supply line to the TDAFW pump. These valves protect the MDAFW pumps in the event of rupture of the steam supply line inside the auxiliary feedwater pump room.

AFW room cooling can be effected by opening the roll up door.

The AFW backup supply of water to the 360,000 gallon (minimum) condensate storage tank, is the 1.5 million gallon City Water Storage Tank, shared with Unit 3.

The CCW system consists of three pumps and two SW cooled heat exchangers. Normally, one or two of the pumps and one heat exchanger are required to supply the necessary cooling to the plant loads. The CCW safety loads are the RHR heat exchangers, the RHR pumps, the safety injection (SI) pumps, the charging pumps and the recirculation pumps (indirectly). The recirculation pumps are cooled by the two motor driven auxiliary component cooling pumps, which reject heat to the CCW loop. The recirculation pumps must be protected during the injection phase, thus cooling is provided by the auxiliary pumps, as CCW is isolated (during the injection phase only) on SI in conjunction with a loss of offsite power.

The containment spray (CS) system contains two CS pumps, which are used in the injection phase. The pumps continue to operate after switchover to recirculation, until the RWST is exhausted. In the recirculation phase, spray is provided by the recirculation pumps or by the RHR pumps, with heat removal provided by the RHR heat exchangers.

The electric power system consists of the six 6.9 kV buses, four 480 V buses, three diesel generators and four 125 V DC buses, two of which serve as a backup. The safety equipment is connected to the 480 V buses, which are normally supplied by 6.9 kV buses 5 and 6. The EDGs require room ventilation for which only two of the EDGs can provide power (at the time of the IPE). Cooling of the EDGs is provided by the essential service water system. Additionally there are three emergency gas turbines providing safety power, each of which can supply the full safety loads of both Units (i.e., IP-2 and IP-3). Two of the turbines (at the time of the IPE) have a blackstart capability. Employment of the gas turbines is modeled as a recovery action in loss of offsite power sequences.

The containment fan cooling system consists of five fan cooler units, with at least three of the five required for a large LOCA if the sprays are inoperable. The fan cooler unit heat exchangers are cooled by service water.

The high pressure safety injection system consists of three safety injection pumps. The SI pumps are intertied such that any primary system cold leg can be supplied by two of the three pumps. Upon receipt of an SI signal, all three pumps start automatically. Pump 22 can be supplied from either of two safety buses (2A or 3A). The other two pumps are supplied by buses 5A and 6A, respectively. These pumps require CCW cooling. The cooling is effected by employing an attached booster pump, mounted on the same shaft as the respective SI pump.

The low pressure injection pumps also serve the RHR function in normal plant operation. The pumps discharge through the RHR heat exchangers and take suction from the RWST or the containment sump. Both pumps start automatically on an SI signal.

The important components of the RCS are the two pressurizer PORVs, the three pressurizer safety valves and the RCP seals. The RCP seals are cooled by CCW and receive injection from the charging pumps (there are three charging pumps). The charging pumps are also cooled by CCW, but have a limited ability to withstand loss of cooling. The onset of charging pump failure due to loss of CCW cooling can be delayed by operating the charging pumps on high speed. The charging pumps also have a hard piped connection to a backup source of cooling, the City Water System. This backup connection must be manually aligned. In case of loss of all seal cooling, the RCP seals are assumed to eventually fail, resulting in a small LOCA.

The recirculation system consists of two dedicated recirculation pumps, drawing suction from the recirculation sump. These pumps are cooled by CCW and also by the booster pumps called the auxiliary component cooling pumps. These latter pumps are used to protect the recirculation pumps in the injection phase, in case of a concurrent loss of offsite power, in which case the CCW pumps are isolated (in the injection phase).

In case the recirculation pumps are unavailable, the recirculation function can be accomplished by the RHR (or low pressure injection) pumps, drawing suction from the containment sump, which is separate from the recirculation sump mentioned above. Both the recirculation and the RHR pumps pass water through the RHR heat exchangers (cooled by CCW), for cooling for core injection and also for containment heat removal. One RHR or one recirculation pump is sufficient to provide both core cooling and the containment heat removal via recirculation spray (see discussion in Sections 2.2.3.1 and 2.2.3.2, dealing with interface issues).

In case that high pressure recirculation is needed, the high pressure (or SI) pumps can be piggybacked on either the RHR or the recirculation pumps.

The service water system consists of two sets of three service water pumps, each set connected to a header. The two headers are designated essential and nonessential. The two headers can be connected but are normally isolated by two closed valves. Either set of pumps can be realigned to provide the essential header, with the other set supplying the non-essential header, which requires manual operator action.

The essential header serves the five containment fan cooler units and the three emergency diesel generators. The nonessential header supplies the two component cooling water heat exchangers and several balance of plant loads. The CCW heat exchangers are not required until the recirculation phase of an accident, at which time flow from at least one of three non-essential pumps is required. Normally, two pumps would be operating on each header. The model conservatively assumes that all pumps are subject to a restart challenge following an initiating event. Simultaneous maintenance of two essential service water pumps is prohibited by technical specifications.

The EDG Building ventilation system consists of five wall mounted exhaust fans, with staggered individual initiation temperature set points. Recently a sixth fan was added, and the whole system was rewired for increased reliability following LOSP (see Section 2.6.2 on improvements). The system also contains pneumatically controlled exhaust dampers (associated with the fans) and four intake louvers, manual intake louvers and electric space heaters. The fan initiation set points range from 80°F to 120°F. The exhaust dampers and intake louvers are also temperature controlled.

The number of fans required during EDG operation depends on the number of EDGs operating and on the outside temperature. Any one fan provide sufficient cooling to support operation of a diesel generator under all but the most severe ambient temperature conditions. Operation of two EDGs requires operation of up to three fans, although the third fan is only required at the upper range of ambient temperatures. The same is true during operation of all three diesels. At the time of the IPE, three of the fans received power from EDG 21, and two of the fans received power from EDG 22, with one of the EDG 21 fans being capable of switching over to EDG 22 automatically, if power to its associated motor control center (MCC) was lost.

Success criteria are described in the accident sequence delineation portion of the report (section 3.1.2). System dependencies are described in a separate section (3.1.5.1), as part of the support system event trees description, and dependency matrices are given.

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. HVAC was determined not to be important except in the emergency diesel generator building, due to a slow rate of temperature rise, frequent surveillance, and simple operator actions to establish air circulation. The motor driven AFW pumps are protected against a steam line rupture in the AFW pump room due to existence of temperature controlled isolation valves in the TDAFW pump steam supply line. Therefore, this failure mode seems not to have been considered.

Tables 3.1-23 and 3.1-24 of the submittal contain the support-on-support and frontline-on-support dependency matrices.

2.2.2 Quantitative Process

2.2.2.1 Quantification of Accident Sequence Frequencies

The IPE used a large event tree/small fault tree approach with a support state event tree to quantify core damage sequences. The event trees are a mixture of functional and systemic top events. The system configuration selected for each top event is dependent on both the initiating event and any support system failures. These system configurations are represented by split fractions which are quantified through use of algebraic system equations. The equations are derived using boolean logic based on the system models developed in the original IPPSS and updated as appropriate.

The system and event tree models were developed and quantified within the RISKMAN risk management software.

The systemic sequences which resulted had a truncation limit of 1.E-10/yr. Mean values were used for initiating event frequencies and conditional split fraction probabilities. An overall core damage uncertainty analysis was also performed using the initiating event uncertainty distributions and the conditional split fraction uncertainty distributions. The IPE took credit for various recovery activities, including the recovery of offsite power. The IPE data used for non-recovery of offsite power are consistent with average industry data cited in an Electric Power Research Institute (EPRI)-sponsored study (NSAC 147).

2.2.2.2 Point Estimates and Uncertainty/Sensitivity Analyses

Mean values were used for the point estimate initiator frequencies and conditional split fraction probabilities. A formal mathematical uncertainty analysis was performed on the results. The submittal reports the point estimate, the mean and the quantiles of the total CDF. Fussell-Vesely importance measures were generated for several important top events.

Several sensitivity analyses were also performed. One analysis raised all human error probabilities (HEPs) to 0.1 to find potentially important sequences which had fallen below the 10^{-7}/yr screening criterion due to the HEPs used. About 50 such sequences were found and the most important of these were discussed. Another sensitivity analysis substituted the old (i.e. IPPSS) RCP seal LOCA model to find the impact on the CDF. The CDF rose by 78%. The third study relaxed the mission time for the turbine driven auxiliary feedwater pump from 24 hours to 8 hours. The reason for considering this time change is that, at 8 hours, RHR entry conditions would be realized given the maximum cooldown rate, and the motor driven AFW pumps may have been repaired. The impact of this change was a reduction in the CDF by 14%.

2.2.2.3 Use of Plant Specific Data

The data collection period was through mid-1991. This means that the data collected for the original IPPSS study in 1982, its amendments in 1982 and 1983 and the PRA Update of 1989 were updated by additional data in the period 1989-1991.

Both demand and time related failures were addressed. Significant Occurrence Reports were the principal source of component failure data. Other sources of plant data were periodic performance tests, daily surveillance reports and turnover sheets, the corporate maintenance tracking system, the central control room and nuclear plant operator log books, maintenance work permits, and LERs.

The plant specific component failure data were used to update generic data with the use of Bayesian techniques. Plant specific data were used exclusively for unavailabilities due to test and maintenance activities.

Plant-specific failure data were gathered for a number of components, including pumps, diesel generators, circuit breakers, check valves, motor operated valves and air operated valves. Table 2 of this review compares the plant specific 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].

IP-2 data are generally in agreement with the NUREG/CR-4550 data, except for the emergency diesel generator failure to start, which is one order of magnitude below the NUREG value.

Table 2. Comparison of Failure Data

Component	IP-2	4550
TDAFW Pump		
fail to start	4.2E-2	3.0E-2
fail to run	2.1E-3	5.0E-3
MDAFW Pump		
fail to start	1.1E-2	3.0E-3
fail to run	9.2E-5	3.0E-5
HHSI Pump		
fail to start	9.7E-3	3.0E-3
fail to run	3.4E-5	3.0E-5
RHR Pump		
fail to start	1.1E-2	3.0E-3
fail to run	4.7E-5	3.0E-5
SWS Pump		
fail to start	1.0E-3	3.0E-3
fail to run	3.0E-5	3.0E-5
CCW Pump		
fail to start	1.0E-2	3.0E-3
fail to run	1.3E-5	3.0E-5

Component	IP-2	4550
IAS Compressor fail to start fail to run	 -- --	 8.0E-2 2.0E-4
Battery Charger Failure	3.7E-6	1.0E-6
Circuit Breaker (480V and 13.8kV) fail to remain closed (spurious open)	7.2E-7	1.0E-6
AC Bus Fault	4.6E-7	1.0E-7
Check Valve (AFW) fail to open	3.8E-5	1.0E-4
MOV Fail on Demand	1.6E-3	3.0E-3
Pressurizer PORV fail to open	3.7E-3	2.0E-3
Emergency Diesel Generator fail to start fail to run	3.1E-3 4.2E-3	3.0E-2 2.0E-3

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

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

2.2.2.4 Use of Generic Data

The source for the generic data was the PLG database. The values used for generic data are not tabulated in the submittal, except where noted for the few components which did not have plant specific data. No major components were assigned generic data for the final value used in the analysis.

2.2.2.5 Common-Cause Quantification

Redundant components were systematically examined to address potential common-cause failures. A plant walkdown was also performed for that purpose. The approach used was the multiple Greek letter approach (MGL). The β and (if applicable) the γ factors are reported in the

submittal, with discrimination based on failure modes (e.g. in general, different values of MGL parameters are given for failure to start as opposed to failure to run). The methodology followed that described in NUREG/CR-4780 ("Procedure for Treating Common Cause Failures in Safety and Reliability Studies"). The cause defense analysis was performed using the methodology described in NUREG/CR-5460 ("A Cause-Defense Approach to the Understanding and Analysis of Common Cause Failures").

A number of categories of components were modeled in the common cause analysis, including: pumps, valves requiring position change, power operated relief valves, check valves, MOVs, AOVs, emergency diesel generators, fans, breakers, batteries, relays and bistables.

A comparison of effective β factors in the submittal vs. those suggested in NUREG/CR-4550 ("reference β factor") is presented in Table 3. NUREG/CR-4550 reports only failure to start β factors. The "effective" β factor above (or the submittal β factor in Table 3) refers to the equivalent β factor which would have been obtained by the licensee had they used the β factor method instead of the MGL approach, such that a comparison is possible between the effective β factor derived from the licensee MGL parameters and the reference β factor, taken from NUREG/CR-4550. In other words, this "equivalent" β factor, when multiplied by the component failure rate, would yield the same probability of common cause failure, for the particular configuration of components, as the MGL approach. For example, for a system of three redundant components, the "equivalent" β factor describes the conditional probability of all three components failing, and is obtained by multiplying the β and the γ factors in the MGL method.

Table 3. Comparison of Common-Cause Failure Factors

Component	Failure Mode	Submittal β factor	Reference β factor
AFW pump (motor driven)	FTS FTR	0.064 0.007	0.056
SWS pump, CCF of 2 pumps	FTS FTR	0.026 0.026	0.026
SWS pump, CCF of 3 pumps	FTS FTR	0.0036 0.0033	0.014
CCW pump, CCF of 2 pumps	FTS FTR	0.054 0.001	0.026
CCW pump, CCF of 3 pumps	FTS FTR	0.0076 6.6E-5	0.014
RHR pump	FTS FTR	0.072 0.01	0.15

Component	Failure Mode	Submittal β factor	Reference β factor
HHSI pump, CCF of 2 pumps	FTS FTR	0.085 0.0074	0.21
HHSI pumps, CCF of 3 pumps	FTS FTR	0.019 0.001	0.10
Containment Spray pump	FTS FTR	0.026 0.04	0.11
MOV, CCF of 2 valves	FTO/FTC	0.01	0.088
MOV, CCF of 3 valves	FTO/FTC	0.0023	0.057
AOV, CCF of 2 valves	FTO/FTC	0.07	0.10
AOV, CCF of 3 valves	FTO/FTC	0.001	0.10
Diesel Generator, CCF of 2 EDGs	FTS FTR	0.007 0.01	0.038
Diesel Generator, CCF of 3 EDGs	FTS FTR	0.001 0.0024	0.018
Pressurizer PORV	FTO	0.07	0.07

The table shows reasonable agreement for two pump systems of various kinds (e.g. AFW, SW, etc.), for failure to start. The equivalent β (derived from the MGL parameters) for the three pump system drops off more precipitously (as compared to the two pump β) in the submittal than in NUREG/CR-4550, resulting in somewhat lower CCF values for that configuration. For instance, in case of the SWS pumps, the submittal β for any two pumps is 0.026 vs. 0.0036 for failure of all three pumps (in a common header), whereas in NUREG/CR-4550, the β drops from 0.026 to 0.014.

Pump failure to run β factors are generally significantly lower than the failure to start β factors (and thus the NUREG/CR-4550 β factors).

For MOVs, the submittal's β factors are significantly lower than the ones reported in NUREG/CR-4550. For AOVs, it should be noted that the reference β of 0.1 was a screening value.

The diesel generator β factors are lower in the submittal, substantially so for failure of 3 out of 3 EDGs. However, the values shown in Table 3 do not include CCFs of some auxiliary systems, which may contribute to EDG CCFs in other plants. These systems are the EDG building fans and the EDG fuel oil transfer pumps. The latter are not needed until one hour into the running time of the associated EDGs. The MGL CCF factors for these components are provided in the submittal.

The general observation is that a few potentially important CCF factors are substantially lower than the ones reported in NUREG/CR-4550. These are the factors for common cause failure of all three HHSI pumps, MOVs and EDGs. In case of diesel generators, it should be noted that additional options are available in case of failure, i.e. the two modeled 100% capacity gas turbines with the blackstart capability. The licensee states that extensive common cause analysis was performed, including a plant walkdown, analysis for applicability of generic events to the IP-2 system configurations, and a cause-defense analysis.

The reason for some low values among the CCF factors used seems to be that the licensee has discarded generic events which do not apply to the IP-2 plant. However, future events, which have not yet shown up in the data base, may be undercounted by this procedure.

2.2.2.6 Initiating Event Frequencies

For plant transients leading to a reactor or turbine trip (including loss of offsite power), plant specific data was collected. The generic database information was then updated with this plant specific information using the Bayesian methodology. For initiating events which were not experienced by the plant such as LOCAs and SGTR, the generic database information was updated by the additional Indian Point 2 experience without such incidents, again using the Bayesian methodology. For the loss of service water, loss of component cooling water, loss of dc bus and loss of 6.9 kV bus initiating events, plant specific models were developed and used to quantify the expected frequency of each event. The ISLOCA analysis "drew substantially upon the work done in NUREG/CR-5102, taking into account the design differences between Indian Point 2 and Indian Point 3, which was one of the plants analyzed in that NUREG" (RAI Responses). Credit is given to the fact that the high/low pressure piping interface is inside the containment, and therefore a certain fraction of interfacing LOCAs will involve pipe breaks inside the containment, where long term inventory control is possible. For the dominant ISLOCA sequence, involving the RHR suction piping, the submittal does not stated what fraction of low pressure piping is inside the containment.

The initiating event frequencies used in the IPE are shown in Table 4.

The initiating event frequencies seem reasonable and are comparable to other PRA studies.

Table 4. Initiating Event Frequencies for Indian Point 2 IPE

Initiating Event	Frequency (/yr)
Large LOCA	2.0E-4
Medium LOCA	4.6E-4
Small LOCA	1.7E-2
Loss of Main Feedwater	1.3
Closure of a MSIV	1.2E-1
Core Power Excursion	1.9E-2
Turbine Trip	1.3
Reactor Trip	2.1
Loss of Offsite Power	6.9E-2
Steam Generator Tube Rupture	1.3E-2
Steam Line Break Inside Containment	4.6E-4
Steam Line Break Outside Containment	5.0E-3
Loss of Service Water	2.5E-3
Loss of Component Cooling Water	8.1E-2
Loss of DC bus D21	2.7E-3
Loss of DC bus D22	2.7E-3
Loss of 6.9 kV bus 2	1.1E-2
Loss of 6.9 kV bus 3	1.1E-2
Reactor Vessel Rupture	3.0E-7
Interfacing Systems LOCA	2.7E-8

2.2.3 Interface Issues

2.2.3.1 Front-End and Back-End Interfaces

Indian Point 2 has both fan cooler and containment spray systems to provide containment cooling functions. During the early phases of an accident, containment spray would be provided from the RWST via 2 dedicated containment spray pumps. Once ECCS recirculation has been

established, containment spray could be provided by either recirculation or RHR pumps. The 2 recirculation pumps draw suction from a dedicated recirculation sump located inside containment that is separate from the containment sump available to the RHR pumps. The RHR pumps are located outside the containment whereas the recirculation pumps are inside the containment.

The IPE assumed that adequate late containment and core cooling can be provided by one RHR pump (or one recirculation pump) and one RHR heat exchanger in scenarios that involve a breach in the reactor coolant system. The licensee states that degradation of heat exchanger performance from fouling should not impact this safety function due to the fact that the Component Cooling Water, a closed system with "tightly controlled chemical specifications" is used for cooling (RAI Responses).

Plant damage states (PDSs) were used to provide the interface between the front- and back-end analyses. Thirteen criteria were used to group the core damage sequences into the PDSs, including the type of initiating event, containment status, ECCS status, RCS pressure, availability of electric power and sequence timing.

The Level 2 analysis is discussed in Section 2.4.

2.2.3.2 Human Factors Interfaces

Based on the limited Fussell-Vesely importance results and descriptions of the most dominant sequences, the following are important human failure probabilities: failure to initiate feed and bleed after a general transient, failure to switch over to recirculation after a LOCA, and failures in timely offsite power recovery or in starting up the gas turbines.

The establishment of containment spray with the recirculation or the RHR pumps requires that the operators carefully balance the water requirements of the sprays versus those of core cooling. The worst case scenario would be to starve the core cooling function. This would be more probable if only one RHR pump or only one recirculation pump were available to provide both the core cooling and the recirculation spray. The success criteria specify that one pump is adequate for both recirculation core injection and recirculation containment heat removal, even if none of the containment fan cooler units are available. Further discussion of the manual actions needed and the rest of the human reliability modeling can be found in Section 2.3 of this TER.

2.2.4 Internal Flooding

The submittal states that internal flooding was addressed in the original Indian Point 2 IPPSS and was not found to be a significant contributor. On this basis the IPE plan proposed revisiting internal flooding as a "coordinated task" during the IPEEE. The approach was accepted by the NRC and therefore was not included in the IPE.

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 internal core damage frequency has a mean of 3.13E-5/yr (this is also the point estimate), and a 5th, 50th and 95th percentile of 1.63E-5, 2.76E-5 and 5.28E-5/yr, respectively. These results thus indicate a very tight distribution for the CDF (an error factor less than 2) which makes the distribution appear questionable. However, an uncertainty analysis was not requested for the IPE process, or in the Generic Letter 88-20, nor is it requested in NUREG-1335. Accident types and initiating events that contributed most to the CDF, and their percent contribution, are listed in Tables 5 and 6.

Twenty one dominant sequences were described in the submittal(four general transients, three losses of offsite power, two losses of support system, eight LOCAs, one LOCA beyond ECCS capability, and three SGTRs). Each of these important sequences has a frequency greater than 1.E-7/yr. These sequences are summarized below in Table 7.

Station blackout contributes 14% to the total CDF. The loss of offsite power contribution is 19%. The ATWS contribution is 5.6%, initiated at >40% power and caused mostly by AFW failures, with some contribution from failure to emergency borate and manually shutdown. The PORV block valve status does not play a significant role (each block valve is closed about 25% of the time). The ISLOCA contribution is 0.1%. The RCP seal failure contribution to core damage sequences appears to be about 5%.

Table 5. Accident Types and Their Contribution to the CDF

Initiating Event Group	Contribution to CDF (/yr)	%
General Transients	1.2E-5	37.3
LOCAs	1.0E-5	33.3
Station Blackout	4.5E-6	14.3
Anticipated Transients Without Scram	1.8E-6	5.8
Loss of Support System Initiating Event	1.3E-6	4.1
Steam Generator Tube Rupture without Stuck Open Relief Valve	1.2E-6	4.0
Steam Generator Tube Rupture with Stuck Open Relief Valve	3.7E-7	1.2
Interfacing System LOCA	2.7E-8	0.09
TOTAL CDF	3.1E-5	100.0

Table 6. Initiating Events and Their Contribution to the CDF[1]

Initiating Event	Contribution to CDF (/yr)	%
Loss of Offsite Power	5.9E-6	19.0
Small LOCA	5.7E-6	18.1
Reactor Trip	4.4E-6	14.0
Loss of Feedwater	2.7E-6	8.7
Turbine Trip	2.7E-6	8.6
Large LOCA	2.6E-6	8.2
Medium LOCA	1.9E-6	6.1

[1]Only the most dominant initiating event contributors to the CDF are listed here.

Table 7. Dominant Core Damage Sequences

Initiating Event	Dominant Subsequent Failures in Sequence	% of CDF
General Transient: Reactor Trip, Turbine Trip, Loss of Normal Feedwater, MSIV Closure or Loss of Primary Flow	Auxiliary Feedwater Fails (mainly due to pump hardware failures) coupled with failures in feed and bleed (usually caused by operator error or failures of PORV/block valves to open.	29.4
Small LOCA	Low Pressure Recirculation failure, usually caused by operator failure to switch over to recirculation.	10.2
Large LOCA	Low Pressure Recirculation failure, usually caused by operator error to realign to the recirculation mode.	5.9
Medium LOCA	Failure of timely post-LOCA cooldown (mainly caused by operator error) and failure in High Pressure Recirculation (mainly caused by operator error to align to the recirculation mode).	3.9
Small LOCA	Failure in RCS injection from the RWST, mainly resulting from failure of the tank outlet valve.	2.8
Small LOCA	Injection phase failure due to failure in HPI (dominated by common cause and random pump failure to start and run) and core cooling recovery failure (dominated by operator error to initiate RCS depressurization).	2.5
Loss of Offsite Power	Failure of EDGs 22 and 23, failure to recover offsite power or power from the gas turbines at ½ hour or at 1 hour (this causes failure of the motor driven AFW pumps as they are powered from buses 3A and 6A, supported by the failed EDGs), coupled with random failures in the turbine driven Auxiliary Feedwater Pump; primary bleed is insufficient due to lack of power to one of the two PORV block valves.	1.8

Initiating Event	Dominant Subsequent Failures in Sequence	% of CDF
Small LOCA	Failure of both ESFAS trains (dominated by hardware failure in one of the logic channels, coincident with the other channel being unavailable during a periodic test).	1.7
Medium LOCA	Inadequate makeup due to failure of any one of the three Accumulators on the intact RCS loops to inject (the medium LOCA Accumulator success criteria may be conservative).	1.6
Steam Generator Tube Rupture	Failure of the early and late cooldown and depressurization (dominated by the common cause failure to open one of the atmospheric relief valves on the intact steam generators; in this sequence, the safety/relief valves on the damaged steam generator will reclose.	1.6
Loss of Offsite Power	Failure of EDGs 21 and 22 causes failure in EDG 23 due to lack of EDG Building Ventilation (this interaction has been fixed by a recent improvement); additionally, the turbine driven Auxiliary Feedwater pump fails randomly, and AC power from offsite or the gas turbines is not recovered at ½ hour or 1 hour.	1.4
Loss of Offsite Power	Failure of EDGs 21 and 22 causes failure in EDG 23 due to lack of EDG Building Ventilation; AC power from offsite or the gas turbines is not recovered at ½ hour, 1 hour or at 3 hours, causing eventual depletion of batteries (however the operators are successful in controlling the TD AFW pump locally using pneumatic instruments); RCP seal LOCA develops due to loss of all seal cooling and core uncovery occurs before AC power can be restored.	1.4

Initiating Event	Dominant Subsequent Failures in Sequence	% of CDF
Steam Generator Tube Rupture	Loss of both ESFAS trains (one train due to hardware problems, the other train due to a periodic test); the relief valve on the damaged steam generator recloses after the RCS has depressurized.	1.3
LOCA beyond ECCS Capacity	None needed.	0.9
Loss of DC Bus 21 or 22	Loss of Auxiliary Feedwater (dominated by common cause and random failures of the motor driven pumps in conjunction with random failures of the turbine driven pump to start and to run); loss of primary bleed due to loss of DC power to one of the two PORVs (both are required).	1.5
Large LOCA	Failure of Low Pressure Safety Injection (dominated by common cause or random failures of both low pressure injection pumps).	0.7
Steam Generator Tube Rupture	Failure of early and late isolation of the ruptured steam generator (dominated by failure to close the AFW Turbine Driven Pump Steam Supply Valves from the ruptured SG and failure of the Safety Relief Valve to close); although late cooldown and depressurization is successful, the RHR fails in the shutdown cooling mode (dominated by failure to open of the hotleg suction valves).	0.7

Table 8 shows the contribution of the dominant systems or functions to the core damage frequency.

Table 8. Top Event Importance

Top Event Description	Failure Importance
Auxiliary Feedwater	0.48
Primary Bleed	0.32
Start Gas Turbine in ½ hr	0.19
Recover Offsite Power in ½ hr	0.19
Low Pressure Recirculation	0.18
EDG No. 22	0.12
Offsite Power Recovery in 1 hr	0.11
Gas Turbine Start and Run in 1 hr	0.10
EDG No. 21	0.10

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 Indian Point 2 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.

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

The IPE indicates that the framework for evaluating the pre-initiator human actions was developed in the original Indian Point Probabilistic Safety Study (IPSSS) completed in 1982 and was continued through the IPE modeling process. The approach reviews pertinent plant procedures related to maintenance, surveillance, and calibration and identifies, during the course of each system analysis, "opportunities for flow path and control switch misalignment and miscalibration errors." In addition, all valves stroked during tests were identified and the potential impact of leaving the valves in the wrong position was evaluated. During the IPE, all procedural changes and changes in aspects such as test intervals were systematically examined to determine their impact on human error quantification. Although not stated explicitly in regard to the pre-initiator analysis, the IPE does state that discussions were held with operations personnel, systems engineers, operations support, and training personnel and that detailed system walkdowns were conducted. Therefore, it appears that relevant information sources were examined and that factors which could influence the probability of human error in pre-initiator actions were considered.

2.3.1.3 Screening Process for Pre-Initiator Human Actions

A screening analysis was not conducted on pre-initiator human actions in the Indian Point 2 IPE. All identified pre-initiator actions received detailed quantification.

2.3.1.4 Quantification of Pre-Initiator Human Actions

The IPE states that the principal source of information for the pre-initiator human actions was the NRC Human Reliability Handbook [NUREG/CR-1278]. Relevant HEPs for restoration faults were taken from the NRC Handbook and both errors of commission and errors of omission were addressed. The type of task (e.g., maintenance vs. inspections), the existence of checkoff provisions in the procedures, and the number of items in the procedure list were factors considered (per NUREG/CR-1278) in determining the HEPs. A special credit was taken for operators checking on Emergency Diesel Generator (EDG) control switch status every four hours through the use of a time dependent unavailability equation. Credit for discovering mispositioned valves during walkarounds was also taken in some cases.

Dependencies between pre-initiator operator actions were identified in situations where actions were carried-out sequentially by the same individuals using the same procedure (e.g., realignment of the isolation valves for both containment spray pumps following test). The NUREG/CR-1278 dependency equations were applied to obtain the relevant HEPs for such situations. In addition, potential miscalibration common cause failures were considered. Complete dependence was assumed between multiple channels of single instrument sets. However, miscalibration across redundant and diverse instrument sets was assumed to be independent. The rationale was that different instruments tend to have different procedures and plant practices indicated that it was unlikely that more than one instrument set would be tested in a single shift. Examples of the treatment of dependencies and common cause failures were provided in response to an NRC Request for Additional Information (RAI).

A review of the 11 HEPs for the restoration events presented in the submittal indicated that the HEPs for restoration events tended to be somewhat lower than what is normally found for similar events in other plants. The lower values were apparently obtained because credit was taken for plant procedures that require double independent verifications. In addition, NUREG/CR-1278 is known to produce somewhat lower values, relative to methods such as the ASEP HRA procedure (NUREG/CR-4772), ostensibly because of the more detailed analysis required by the method.

Limited discussion of miscalibration events was provided in the submittal. Miscalibration events were discussed in more detail in the licensee's response to an NRC RAI, but the actual quantification approach was still not sufficiently documented. The response to the RAI states that the failure of a single instrument loop to provide an actuation signal due to miscalibration was determined to be 3.6E-4. While this is not an unreasonable value, exactly how it was determined was not explained. This value could not be found in Table 3.3-5 of the submittal, which purportedly presents the criteria and values for pre-initiator events. In addition, a listing of all the miscalibration events was not provided and it was unclear exactly how many were actually modeled. Treatment of miscalibration events could be a minor weakness of the pre-initiator analysis.

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 licensee's 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 Indian Point 2 IPE did not make an explicit distinction between response and recovery type post-initiator human actions as defined above. However, the submittal states that no credit was taken for recovery actions unless they were supported by written operating procedures and were demonstrated to be viable. Given this assertion, the submittal addresses and quantifies all post-initiator human actions with the same approach and methodology. (The only exception involved recovery of offsite power which is normally based on actual industry data as opposed to HRA

analysis). The adequacy of procedures and training were considered in determining human actions and their HEPs.

In response to an NRC RAI, it was stated that all but one operator action was modeled in the front line system event trees (as opposed to the support system event trees). This was possible in part because with the exception of manual scram, manual backups to failed automatic actions were conservatively not modeled. The IPE submittal noted that this facilitates consideration of dependencies, which is true. However, by not taking credit for manual recoveries of failed automatic actions, the results of the IPE could be somewhat distorted. To assume that such recoveries would fail may be conservative, but is not realistic. At a minimum, such an approach prevents the licensee from getting the full benefits of the IPE and may impact discovery of issues related to accident management.

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

The submittal states that operator actions already addressed in the IPPSS (completed in 1982) were reevaluated using "today's human reliability techniques and data," while accounting for current plant procedures and training. In addition, "new operator actions were added to the front-end model to reflect current reactor safety issues." The submittal also notes the importance of HRA "issues" such as task analyses relating to current EOPs and the use of plant specific data from simulator training. In the executive summary it notes that additional steps were taken to verify that original modeling was representative of the current plant and that available simulator runs were a source of information. The general methodology section notes that in developing the accident sequences, operator actions expected to occur in accordance with the EOPs were carefully considered and integrated into the analysis. A number of simulator observations, timings, and procedure walkthroughs were made.

2.3.2.3 Screening Process for Post-Initiator Response Actions

A screening analysis for post-initiator response type actions was not performed in the Indian Point 2 IPE. Operator actions were reviewed in the context of the accident sequences to which they would contribute (in order to assess dependencies and context effects) and detailed quantification was performed for each action.

2.3.2.4 Quantification of Post-Initiator Human Actions

Quantification of post-initiator human actions was based primarily on the EPRI *Generalized Event Tree Representation of Post-Accident Actions* [EPRI NP-6937], in conjunction with EPRIs operator reliability experiments (ORE) based methodology [EPRI NP-6560-L]. Three parameters are used in the quantification process: the probability of unrecovered detection errors (p_1), the probability of non-response within available time (p_2), and the probability of failure to correctly execute the appropriate step in a procedure (p_3). The method selects some base probabilities (e.g., for the response execution parameter (p_3)) from the NRC Handbook [NUREG/CR-1278]. In addition, in the licensee's response to the NRC RAI, it was stated that for cases in which "the crew were not on

a time critical path," a nominal value for probability of diagnosis (p_2) of 2.7E-3 was applied based on the ASEP HRA methodology [NUREG/CR-4772]. This approach is discussed in more detail below.

The first parameter addressed by the model (p_1), was assumed to make a negligible contribution to total failure probability on the basis of a detailed control room design review and numerous human factors improvements. Such an assumption is consistent with that of other methodologies.

The p_2 parameter includes the potential for operator diagnosis failure and explicitly considers the impact of time on the potential for correct diagnosis and response initiation. Time was usually addressed through the use of the Human Cognitive Reliability/Operator Reliability Experiments (HCR/ORE) time reliability correlation (TRC) (i.e., EPRI NP-6560-L), which is a modification of the Human Cognitive Reliability (HCR) correlation [NUREG/CR-4835]. The modification is based on results from the ORE study. The TRC equation produces failure probabilities for p_2 as function of time and as a function of several performance shaping factors (PSFs). In addition (and as noted above), in the plant response to an NRC RAI it is noted that the p_2 parameter is assigned a nominal failure probability of 2.7E-3 for diagnosis when the time available for diagnosis exceeds 60 minutes and the time needed to respond was expected to be less than 15 minutes. This nominal failure probability was also assigned to operator actions designated as "immediate actions" in the EOPs, which can be assumed to be performed from memory. While it was stated that the 2.7E-3 value was applied on the basis of the ASEP HRA methodology [NUREG/CR-4772], no basis for the choice of the value (which is not unreasonable) was provided.

The p_3 parameter includes factors for recovery of initial implementation (response execution) failure if sufficient time is available, and thus also considers the impact of time on failure probability. This parameter is primarily concerned with "slips" in execution of the response (errors of omission or commission) and the initial HEPs used for this parameter were selected from the NRC Handbook (NUREG/CR-1278). The submittal notes that the most common value for the initial HEP for p_3 was 1.2E-3.

According to a set of rules based essentially on the relative amounts of diagnosis time available and the assumption that crews will begin to take action well within the available time window, recovery factors of either 0.3 or 0.1 could be applied to the initial HEP for p_3. The recovery values reflect the amount of time presumed to be "leftover" for detecting execution errors. The rules and recovery values were assumed to be reasonable on the basis of expert judgment. To obtain the total failure probability for an event, the HEPs for p_2 and p_3 are summed.

2.3.2.4.1 Estimates and Consideration of Operator Response Time

The Indian Point 2 IPE used several sources to determine the total time available to recognize the need for a required action and to perform it (T_w). The sources used for estimating the available time window included: 1) generic NSSS vendor analyses, 2) accident sequence chronologies from MAAP calculations, and 3) other types of thermal-hydraulic analyses. In addition to estimates of the available time window to diagnose and conduct the needed action, the HCR/ORE TRC equation

(which directly gives the probability of failure for the operator diagnosis (p_2)), requires estimates of several other parameters. These include task implementation time (T_a), "crew median response time" ($T_{1/2}$), and the time available for diagnosis ($T_{w - cr}$), which is given by the difference between total available time and task implementation time. Estimates of task implementation times were based on observations of simulator training scenarios, job performance measures, or expert judgment. Estimates of crew median response times were typically based on insights from simulator training, discussions with plant staff, and information provided in other PRA studies.

The HCR/ORE TRC as discussed in the submittal is:

$$P(T_{w\text{-}cr}) = 1 - \Phi[\ln(T_{w\text{-}cr}/T_{1/2})/\sigma]$$

where "w - cr" stands for the time window for the cognitive response (diagnosis), Φ is the standard normal cumulative distribution, and σ is the standard deviation. With an estimate of the standard deviation, the equation gives the probability of the crew failing to diagnose the needed actions and initiating the response in the time available (i.e., p_2). As is discussed in the next section, the process used to provide an estimate of the standard deviation allows for the application of several PSFs relevant to a successful diagnosis.

2.3.2.4.2 Other Performance Shaping Factors Considered

An important assumption of the p_2 model, is that the standard deviation will be related to how demanding a particular accident scenario is for a crew. Therefore, a large value for σ will be indicative of a difficult diagnosis, while a small value will reflect an expected consistency in crew response due to the relative ease of the diagnosis. Determination of the estimate for σ was based on the use of a decision tree which considers the impact of PSFs such as procedural guidance, adequacy of alarms, complexity, training, implementation difficulty, and stress level. Expert judgment is used to work through the tree. The basis for the decision tree endpoints (values for σ) was indicated as being derived from results of the EPRI-ORE program, "insights from practical HRA", and insights from observed simulator training. Thus, the method attempts to consider appropriate PSFs, which is good, but evidence supporting the validity of the decision tree was not provided.

2.3.2.4.3 Consideration of Dependencies

As discussed above, dependencies related to the impact of time on the crew's ability to complete important actions and to recover failures were clearly addressed in the IPE.

Another type of dependence concerns the extent to which the failure probabilities of multiple human actions within a sequence 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 sequence or cut set. While the same basic action can be asked in a number of different sequences, different contexts can obviously lead to different likelihoods of success.

Dependence among multiple human actions was addressed in the IPE. The submittal indicates that each of the human actions in the system models were reviewed in the context of the accident sequences to which they would contribute, in order to determine if any of those actions were dependent. A set of simple guidelines was used to determine the extent to which particular actions were dependent, with either complete or zero dependence being assigned. The process essentially involved identifying (in sequential order) all of the top events in the frontline system event trees and their associated operator actions and using the guidelines to assess dependence. Examples of the process were provided in a response to an NRC RAI and the results appeared reasonable. The guidelines, as presented in the response to the RAI, were as follows:

a) If two human action events are associated with responses to the same plant symptoms, and the responses are required within the same time frame, the cognitive part of the failure probabilities are considered to be totally dependent unless one of the actions is an immediate action response step (see (e) below).

b) As a corollary to this, if in the chronological development of the scenarios, a demand for a human action follows a successful action, and the procedural instructions for both events are closely related, the cognitive failure probability of the second HEP should be very small and can be neglected (i.e., success in the first event implies a successful recognition of the scenario).

c) If the redundant human actions are credited in different time phases of the accident development such that independent diagnosis is possible (e.g., after one hour the technical support center is manned), the actions may be considered independent.

d) If redundant human actions are in response to diverse cues, the actions may be regarded as independent.

e) All actions taken in response to "immediate action steps" performed from memory (e.g., E-0, Steps 1-4 and ES-0.1, Step 1), are considered to be independent from other actions.

Thus, the factors considered included similarity of symptoms/diversity of cues, time proximity, chronological development, and whether or not the actions could be considered immediate action steps performed from memory. In addition, descriptions of the post-initiator human actions and discussions of the quantification process indicated that context effects were considered.

2.3.2.4.4 Quantification of Recovery Type Actions

As discussed above, a special analysis of recovery type actions was not conducted. All post-initiator human actions in the models were analyzed in the same manner, with the extent and adequacy of procedures treated as a PSF.

2.3.2.4.5 Human Actions in the Flooding Analysis

The submittal states that internal flooding was addressed in the original Indian Point 2 IPPSS and was not found to be a significant contributor. On this basis the IPE plan proposed revisiting internal flooding as a "coordinated task" during the IPEEE. The approach was accepted by the NRC and therefore was not included in the IPE.

2.3.2.4.6 Human Actions in the Level 2 Analysis

The back-end section in the IPE notes that events in the containment event tree (CET) may represent phenomenological processes, operator actions, or system failure resulting from severe accident phenomena. It also notes that since these events are different in character, the quantification process must recognize these differences. However, no further discussion of back-end related operator actions could be found. Except for the recovery of ac power for the SBO sequences, operator recovery actions are not included in the CET structure in the Indian Point 2 IPE.

2.3.2.5 Important Human Actions

A list of the more important human actions was not provided in the IPE. However, in discussing the dominant sequences and in a summary of core damage sequences whose frequencies would have been above 1.0E-7 if the human interactions included in them were less reliable (i.e., set to 0.1), several important operator actions were noted and are briefly described below in Table 9, along with their HEP values.

The following appeared to be the most important human actions: failure to initiate feed and bleed after a general transient, failure to switch over to recirculation after a LOCA, and failures in timely offsite power recovery or in starting up the gas turbines.

In addition, in the licensee's response to an RAI, one important action listed in Table 9 below is discussed in detail. It is noted that the establishment of containment spray with the recirculation or the RHR pumps requires that the operators carefully balance the water requirements of the sprays versus those of core cooling. This involves a series of manual actions upon switchover to recirculation. The potential for the operator to fail to perform these actions is accounted for in the human reliability analysis associated with the switchover. The establishment of adequate flow to the core is actually specified at two separate points in the EOP governing the switchover. The operator is first asked to verify that a specific minimum flow rate, representing adequate flow, exists as one of the specific switchover steps. Several steps later, in the same procedure, the operator is again directed to assure that the minimum specified core flow rate is established and that the core

spray flow also be verified. "It is highly unlikely, given the operator has correctly responded to the switch over cue, that he would fail to respond to at least one of the two directives" (RAI Responses).

Table 9. Important Operator Actions

Event Description	Human Error Probability (HEP)
Operator fails to provide timely primary feed and bleed in transient sequences with loss of secondary side cooling.	7.1E-3
Operator fails to re-align to recirculation after post LOCA cooldown. This is important in small and medium LOCAs where RCS makeup is lost when the RWST is depleted.	1.2E-4
Operator fails to initiate post LOCA cooldown in time to permit low pressure recirculation upon exhaustion of the RWST and high pressure recirculation failure.	0.1
Operators fail to correctly realign to the recirculation mode and HP recirculation fails. This action involves the inadvertent switching of the HP injection pumps off at the conclusion of the recirculation switch over procedure and failure to recover the pumps.	2.3E-5
Operator fails to initiate core cooling recovery which occurs due to failure to initiate RCS depressurization in a timely manner. This is important in small or medium LOCAs with HPSI failed.	5.6E-2
Operator fails to reset the MCCs which supply the EDG fuel pumps. This is important in LOSP sequences with failure of all three EDGs due to loss of the fuel oil supply to the EDG day tanks, failure to reset the MCCS to the fuel pumps and, failure to recover power.	3.72E-5
Operator fails to isolate ruptured SG. This is important in SGTR sequences.	1.0E-5
Operator fails to initiate containment sprays in general transients after failure of AFW and MFW.	1.2E-4
Operator fails to perform manual trip within ten minutes after auto trip fails. (ATWS)	4.0E-3
Operator fails to start gas turbines. This is factored into the offsite power non-recovery probabilities and is likely important in LOSP and SBO sequences.	N/A

2.4 Back End Technical Review

2.4.1 Containment Analysis/Characterization

2.4.1.1 Front-end Back-end Dependencies

The interface between the front-end and back-end analyses consists of a set of plant damage states (PDSs). PDS definition is discussed in Section 3.1.6 of the IPE submittal. PDSs are defined by a logic diagram (an event tree) with the following headings:

- Containment bypass,
- Initiating event,
- Power Recovery,
- Containment spray status,
- Containment heat removal status,
- RCS pressure at core damage and vessel failure, and
- Status of in-vessel injection.

Similar to NUREG-1150, four RCS pressure ranges are considered in the Indian Point 2 IPE: high-high pressure, about the pressurizer relief valve setpoint pressure or greater than 2,350 psig; high pressure, between 2,000 to 2,350 psig; low-high pressure, between 200 to 2,000 psig; and low-low pressure, less than 200 psig. Although the heading is described above as "RCS pressure at core damage and vessel failure", it is based on the RCS pressure that can be determined from the Level 1 sequence description, RCS depressurization between core damage and vessel breach by hot leg creep rupture is considered in the IPE for CET quantification.

In the Indian Point 2 IPE, the RCS pressure for a PDS depends on the initiating event and the depressurization mechanisms developed during accident progression before core damage. Only large LOCA events are assigned to PDSs with the low-low RCS pressure range and both medium and small LOCA events are assigned to PDSs with a low-high pressure range. Transient and SBO events are assigned to PDSs with a high-high pressure range if no leak is developed and to PDSs with a high or low-high pressure range if a leak area is developed by either a stuck-open valve or by a transient induced LOCA. In the Indian Point 2 IPE, PDSs with a high-high pressure range contribute to about 50% of total CDF and PDSs with a low pressure range contribute to only about 10% of total CDF.

Three containment bypass PDSs are defined in the IPE. Two of the bypass PDSs involve steam generator tube rupture (SGTR) and the third bypass PDS involves interfacing system LOCA (ISLOCA). The difference between the two SGTR PDSs is the status of the secondary side valves. For one SGTR PDS, a valve sticks open, causing a continuous and more severe release, while for the other SGTR PDS, the valves open intermittently to relieve the pressure and close when the RCS pressure drops below their setpoint pressure, resulting in the release being terminated and thus a less severe release. The assignment of an SGTR initiated event to either of the above SGTR PDSs depends on whether system cooldown and depressurization is successful during Level 1 accident

progression, based on parameter values that can be determined from the Level 1 analysis. The three bypass PDSs contributes to about 5% of the CDF; most of which is from SGTR sequences. The contribution from ISLOCA is less than 0.1%.

Fifty four (54) PDSs are defined from the Level 1 analysis. After combination of similar PDSs and elimination of PDSs with zero CDF, thirty one (31) PDSs remain for use in the level 2 analysis. In some of the combinations, more severe PDSs are assigned to less severe PDSs. For example, in Figure 3.1-11 PDS 7, where containment sprays are not available, is assigned to PDS 1, where containment sprays are available. However, because the receiving PDSs have significantly higher frequencies than the donor PDSs (by about two orders of magnitude), the combination method used in the IPE appears adequate.

The most probable PDS obtained from the PDS definition is PDS 11, which involves a general transient with the RCS at high-high pressure, with no in-vessel injection, and with both containment spray and containment heat removal available (39% of CDF). This is followed by a PDS with a LOCA initiator, with the RCS at low-high pressure, without in-vessel injection and containment spray, but with containment heat removal available (18% of CDF).

The PDSs defined by the logic diagram described in the Indian Point 2 IPE submittal are of sufficient detail to provide a proper account of the front-end and back-end dependencies and adequate information for back-end accident progression analysis.

2.4.1.2 Containment Event Tree Development

Quantification of severe accident progression is performed using containment event trees (CETs). Decomposition event trees (DETs) are used to quantify the top events of the CET. The development of the CET is discussed in Sections 4.5 of the IPE submittal. Except for PDSs associated with containment bypass sequences, a general containment event tree structure is used for all other PDSs. The general CET includes the following top events:

1. Mode of induced primary system failure (e.g., hot leg failure or induced SGTR),
2. Debris cooled in-vessel,
3. Loss of isolation or mode of early containment failure,
4. Containment heat removal or recirculation spray available early,
5. Debris cooled ex-vessel,
6. Mode of late containment failure,
7. Recirculation spray available late,
8. Containment failure long term.

The CETs for the bypass PDSs have only one branch to address the bypass status. In general, the CETs 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. Figures 4.5-1 through 4.5-31 of the submittal show the event trees for the thirty one

PDSs. Also shown in these figures are the split fractions for the event tree branches and the conditional probabilities of the CET end points (or source term categories).

Discussions of DET development and quantification are provided in Section 4.6 of the submittal. Figures 4.6-1 through 4.6-8 show the DETs for the eight top events of the general CET. The DETs include the severe accident phenomena and the containment events that are important to accident progression. Although all the important severe accident containment failure modes that are discussed in NUREG-1335 are addressed in the Indian Point IPE, some of them are dismissed in the IPE after brief discussions in Section 4.4 of the submittal and therefore are not included in the DETs. These include those associated with ex-vessel steam explosion, vessel thrust force, hydrogen burn before vessel breach, global hydrogen detonation, and failure of containment building penetrations. The exclusion of these failure modes from further evaluation is primarily based on results from the NUREG-1150 studies. Additional discussions on this issue are provided in the licensee's response to the RAI. Except for the above phenomena, the DETs include all the severe accident phenomena and containment events that are important for accident progression.

In addition to the eight DETs used for the general CET, a DET is also used in the IPE to determine the fission product retention effectiveness of the Auxiliary Building for the ISLOCA PDS. This is investigated for the ISLOCA PDS because the most likely location of ISLOCA failure would be into the Auxiliary Building, and, according to the IPE, the release would depend on whether the break location is submerged. An investigation of the Auxiliary Building configuration shows that water cannot accumulate, and as a result, fission product retention in the Auxiliary Building for the ISLOCA PDS is neglected in the IPE. No credit for the decontamination effect of the Auxiliary Building is give for the other PDSs either.

The DETs used in the Indian Point 2 IPE are logical and of sufficient detail. Their quantification is primarily based on the results obtained in the NUREG-1150 studies. This is supplemented by the data obtained from plant-specific MAAP analyses, and in some case, by the engineering judgment of the analyst. In general, the quantification process for the CET and the associated DETs is systematic and traceable. Although the values assigned in the IPE are in general adequate, their adequacy cannot be verified because of the limited scope of this technical evaluation. Some items that are of interest are discussed in the following.

Modes of Induced Primary System Failure and RCS Depressurization

The modes of induced primary system failure considered in the CET include induced hot leg failure and SGTR failure. Only hot leg failure is considered to have a sufficient break size to cause significant RCS depressurization. Induced SGTR will not result in significant RCS depressurization but will result in a containment bypass. RCS depressurization by operator actions is not modeled in the CET structure.

The values used for the determination of induced SGTR failure and induced hot leg failure are those from the NUREG-1150 study. Induced SGTR is assumed to be possible only if the RCS pressure is in the high-high range (0.018 conditional probability). Induced hot leg failure is assumed to be

possible when the RCS pressure is in either the high-high range (0.72 conditional probability) or the high range (0.034 conditional probability).

A review of the CET and its supporting DETs shows that although RCS depressurization due to hot leg failure is evaluated in the CET, its effect on RCS pressure is not modeled in the DET for the determination of early containment failure. On the other hand, its effect is modeled in DETs for the determination of the availability of containment heat removal and containment spray, in-vessel debris cooling, and ex-vessel debris coolability due to debris dispersion.

The omission of RCS depressurization due to hot leg failure may have either a positive or negative effect on early containment failure, depending on what containment failure mechanism is dominant in the CET. While the probability of containment failure due to in-vessel steam explosion (or alpha mode failure) increases with RCS depressurization, the probability of containment failure due to the phenomena associated with high pressure melt ejection (HPME) decreases with RCS depressurization. Since the contributions to early containment failure from both of the above containment failure mechanisms are small in the IPE, the omission of RCS depressurization due to hot leg failure does not seem to have a significant effect on the containment failure profile of Indian Point 2.

Early Containment Failure

Early containment failure is defined in the IPE as shortly before, at, or soon after reactor vessel failure. The mechanisms that are considered in the DET for early containment failure include in-vessel steam explosion (alpha mode failure) and containment pressurization immediately after vessel breach. Although the probability of alpha mode failure for the Indian Point 2 IPE is based on the data and the supporting information used in the NUREG-1150 study, the actual values used in the IPE for the alpha mode failure calculation are much smaller than those used in the NUREG-1150 analyses. The probability values used in the Indian Point 2 IPE for low and high RCS pressure are 4E-5 and 4E-6, respectively. Corresponding values of 8E-3 and 8E-4, respectively, are used in NUREG-1150.

As in the NUREG-1150 study, the containment pressure rise at vessel breach is treated as a single issue in the IPE, including contributions from all the phenomena associated with vessel breach (e.g., DCH, hydrogen burn, etc.). The pressure rise rate is also based on the data used in the NUREG-1150 study.

Debris Cooled Ex-Vessel and Long-Term Containment Failure

The probability values assigned in the IPE for debris cooling depend on the depth of the debris pool and the availability of cooling water. Three debris pool depths are used in the IPE for debris coolability assessment: a deep pool with depth greater than 25 cm, a shallow pool with depth between 10 and 25 cm, and a very shallow pool with depth less than 10 cm. The probability of forming a deep debris pool in the reactor cavity is estimated in the IPE based on simple calculations (of the value of core mass and the spread area in the cavity) and engineering judgment. With

cooling water, the probability of debris cooling is assumed to be 0.5 (uncertain) for a deep debris pool and 1.0 (cooling assured) for a shallow or a very shallow debris pool. Without cooling water, debris cooling is assumed to be impossible for a deep or a shallow debris pool and likely (0.9) for a very shallow debris pool.

The probability of long-term containment failure (by basemat melt-through) given core debris not cooled ex-vessel is assigned a value of 0.25 in the IPE. This value is based on the NUREG-1150 value for Surry. The Surry value is used because the composition of the concrete used in the basemat for Indian Point 2 is closer to that used in Surry than in Zion.

Late Containment Failure

The mechanisms considered in the IPE for late containment failure include containment over-pressurization and basemat melt-through. The primary cause of late containment failure is from steam overpressurization, due to the loss of containment heat removal capability. The probability of containment failure due to a late burn of combustible gases is also considered in the IPE. This failure mode is considered to be possible only if the debris is not cooled ex-vessel and has a small contribution to the late containment failure probability.

Recovery Actions

No recovery actions are considered in the CET. AC power recovery is a PDS parameter and is considered only for the SBO sequences. The recovery actions that are considered in some other IPEs, such as RCS depressurization by operator actions and recovery of containment heat removal capability, are not modeled in the CET structure of the Indian Point 2 IPE. The lack of consideration of recovery actions in the CET can be considered to be conservative, but limits the capability of the CET as a tool to investigate the effectiveness of operator recovery actions on containment performance for accident management strategy development.

2.4.1.3 Containment Failure Modes and Timing

The Indian Point 2 containment failure characterization is described in Section 4.2.2 of the IPE submittal. The median containment failure pressure used in the IPE is 126 psig. This is based on the containment capability analysis performed by the United Engineers for Indian Point 2 [United Engineers]. The IPE notes that the result from the United Engineers analysis for Indian Point 2 as well as the Stone & Webster analysis performed for Surry, is used in the NUREG-1150 study to provide the basis for the derivation of the containment failure pressure and failure sizes for Surry [NUREG/CR-4551]. To provide additional support for the selection of the containment failure pressure, the IPE submittal also discusses the containment failure pressures obtained for Indian Point 2 in the reports prepared by Sandia and Los Alamos, both of which show lower but similar values. Because the Indian Point 2 data was used in the development of the failure size distribution for Surry, the distributions for the failure sizes developed in NUREG-1150 for Surry are considered in the submittal as directly applicable to Indian Point 2.

2.4.1.4 Containment Isolation Failure

Containment isolation failure is not considered in the IPE as a PDS parameter, but assessed in the CET. Since containment isolation is not strongly dependent on the other systems considered in the CETs, it is evaluated independently from the PDSs, and a constant probability of 0.0005 is used for all PDSs. This value is similar to that used in NUREG-1150 for the Surry analysis.

The treatment of containment isolation in the IPE is briefly discussed in Section 4.6.3.3 of the submittal. A more detailed discussion was provided in the licensee's response to the RAI. The analysis of containment isolation failure in the IPE includes identification and screening of containment penetrations, consideration of the signals for the containment isolation system, and the quantification of the containment failure mode including the consideration of common cause failure. According to the IPE, the existence of an undiscovered containment leak of any significance at Indian Point 2 is highly unlikely because of the use of a Weld Channel and Penetration Pressurization system (WCPPS), and leakage from any portion of the system is alarmed in the central control room. The Indian Point 2 design also incorporates an Isolation Valve Seal Water System (IVSWS). This system will provide an immediate indication of failure of either of the redundant valves to close. Because of the above systems, it is assumed in the IPE that both the probability of pre-existing leakage path and the contribution from a failure of the initiation signal are negligible. The calculated frequency of isolation failure for Indian Point 2 is 3.5E-4. A value of 5.0E-4 is used in the IPE.

From the description provided in the IPE submittal and the licensee's response to the RAI, it seems that the analyses have addressed all five areas identified in the Generic Letter regarding containment isolation.

2.4.1.5 System/Human Responses

Except for the recovery of ac power for the SBO sequences, operator recovery actions are not included in the CET structure in the Indian Point 2 IPE.

For ac power recovery, a time dependent model is used. The model accounts for possible restoration of offsite power or the recovery of power using the procedures in place to start and load the gas turbines. At Indian Point 2, the Loss of AC Power procedure directs the operators to restore power by restarting an Emergency Diesel Generator (EDG), and if unsuccessful, starting a Gas Turbine. However, in the IPE, the recovery of the EDG is not credited in the analysis. Quantification of gas turbine recovery is based on plant specific data, and quantification of restoring offsite power is based on NSAC-166, "Losses of Offsite Power at U. S. Nuclear Power Plants". According to the IPE submittal, a comparison of NSAC-166 with Indian Point plant specific data shows reasonable consistency, and it is used in the IPE because it includes more recent information.

2.4.1.6 Radionuclide Release Characterization

The radionuclide release characterization is described in Section 4.7 of the IPE submittal. The end states of the CETs are the source term categories. Seven parameters are used to define a source term category. They are:

1. Containment bypass,
2. Debris cooled in-vessel,
3. Alpha mode failure,
4. Containment isolation status,
5. Time of loss of containment pressure boundary integrity,
6. Time recirculation sprays operate,
7. Mode of loss of containment pressure boundary integrity.

An event tree structure is presented in the submittal to show the definition of the source term categories and their frequencies. A total of 26 source term categories (STCs) are defined in the IPE. Release fractions for the STCs are determined via the analyses of representative sequences using the MAAP computer code.

In the Indian Point 2 IPE, the 26 release categories are further grouped into five release category types. This grouping is based on the iodine and cesium release fractions to facilitate comparison with GL 88-20 sequence selection criteria in terms of release magnitudes. Among the five release types, Types I and II have release fractions for iodine and cesium greater than 0.2 and 0.04, respectively. The release magnitudes of both Type I and Type II releases exceed those defined in WASH 1400 for the PWR-4 release category. According to the IPE results, the primary contributors to Type I releases are an SGTR with stuck-open valve (1.2% of CDF) and an ISLOCA (less than 0.1% total CDF). Since release timing is not considered in the definition of the release types, Type II releases include release categories with both early and late release timing. The leading contributor to Type II release with an early release timing is SGTR without a stuck-open valve (4.9% of CDF). Most of this is from SGTR as an initiating event (4% of CDF) and the remainder is from ISGTR (0.9% of CDF).

Generic Letter 88-20 states that "any functional sequence that has a core damage frequency greater than or equal to 1E-6 per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release categories of WASH-1400," should be reported by the IPEs. The IPE submittal appears to fulfill this request.

2.4.2 Accident Progression and Containment Performance Analysis

2.4.2.1 Severe Accident Progression

The modular Accident Analysis Program (MAAP) was used in the Indian Point 2 IPE for accident progression analysis. The MAAP analyses were performed to assist in the quantification of the CET and for estimating the source terms. As mentioned above, the CET quantification for Indian Point

2 is primarily based on results from NUREG-1150 work, particularly those from the Surry analysis. MAAP calculations are used in the IPE primarily for source term definition. According to the submittal, default values were used for all MAAP model parameters except that associated with the in-core blockage model (which was deactivated). Detailed information on the MAAP code model used in the IPE is not provided in the submittal. Uncertainties on the parameter values and sensitivity of sequence analysis results to phenomenological uncertainties are also not discussed in the IPE submittal.

MAAP calculations were performed in the IPE for eight sequences, and release fractions for the 26 source term categories are characterized by their similarity to one of the eight MAAP calculations. The sequences selected for MAAP calculations include a V sequence, two SGTR sequences with and without a stuck-open valve, two SBO sequences with early containment rupture or leak, and three small LOCA sequences with late containment failure and with different core injection and containment spray availability status. Detailed discussions of how these sequences were selected to represent the source term categories are not provided in the submittal, and, except for release fractions, results of the MAAP calculations are not provided in the submittal. Nonetheless, the selected sequences seem to provide a reasonable representation of the source term categories. Furthermore, in the IPE submittal, the release fractions obtained from the MAAP calculations are compared with those obtained from previous studies (e.g., the NUREG-1150 study for Surry). The comparison shows reasonable agreement.

2.4.2.2 Dominant Contributors: Consistency with IPE Insights

Source term categories (or containment failure modes) and their frequencies obtained from the Indian Point 2 CET quantification are discussed in Section 4.7 of the submittal. Table 10, below, shows a comparison of the conditional probabilities for the various containment failure modes obtained from the Indian Point 2 IPE with those obtained from the Surry and Zion NUREG-1150 analyses.

Table 10. Containment Failure as a Percentage of Total CDF

Containment Failure Mode	Indian Point 2	Surry	Zion
Early Failure	0.13	0.7	1.4
Late Failure	9.0	5.9	24.0
Bypass	6.2	12.2	0.7
Isolation Failure	0.05	*	**
Intact	84.6	81.2	73.0
CDF (1/ry)	3.1E-5	4.0E-5	3.4E-4

* Included in Early Failure, approximately 0.02%
** Included in Early Failure, approximately 0.5%

According to the Indian Point 2 IPE, containment bypass is the dominant contributor to environmental release. Of the 0.062 conditional probability of containment bypass, 79% (or 4.9% of CDF) is from SGTR without a stuck-open valve, 19% is from SGTR with a stuck-open valve, and about 2 % is from ISLOCAs (less than 0.1% of CDF). For the SGTR sequences, environmental release is more severe if the safety valves on the secondary side fail to reclose and fission product release to the environment continues after the RCS is depressurized to a pressure below the valve setpoint. On the other hand, a smaller release is expected if the valves reclose successfully. In the Indian Point 2 IPE, induced SGTR is assigned to the category of SGTR without a stuck-open valve, contributing about 17% to this category.

The conditional probability of early containment failure for Indian Point 2 is only 0.13% of total CDF. About half of this is from SBO sequences and another half from the other transient initiated sequences. The contribution to early containment failure from LOCA sequences is negligible. Since the total CDF of SBO is only about one third of that of other transients, the conditional probability of early containment failure for SBO sequences is about three times that for other transients. The contribution to early containment failure due to alpha mode failure is very small (about 5E-6 of CDF). This is because of the small conditional failure probability values assigned in the IPE for this failure mode (4E-5 for low pressure and 4.E-6 for high pressure).

The conditional probability of late containment failure for Indian Point 2 is about 9% of total CDF. Of the 9% conditional probability, 3.5% is from LOCA sequences, 3% is from transient sequences, and 2.5% is from SBO sequences. However, on a conditional basis, it is more likely for an SBO sequence to develop a late failure (about 16% of SBO CDF) than LOCA sequences (about 12% of LOCA CDF) and other transient sequences (about 6% of transient CDF).

2.4.2.3 Characterization of Containment Performance

As shown in Table 10, for Indian Point 2, the core damage frequency (CDF) is lower than that reported in the NUREG-1150 studies for Surry and Zion, and the containment failure profile is in general consistent with those obtained in the NUREG-1150 studies for Surry and Zion. The conditional probability of early containment failure found in the Indian Point 2 IPE (0.13% of total CDF) is less than that calculated for Surry and Zion in NUREG-1150. Although part of this may be attributable to the lower probability of containment failure due to overpressure at vessel breach, a significant part may also be attributable to the low probability values assigned in the Indian Point IPE for alpha mode failure.

The probability values for alpha mode failure assigned in the NUREG-1150 analyses for Surry and Zion are 8E-3 (or 0.8%) for low RCS pressure and 8E-4 (or 0.08%) for high RCS pressure. Because there usually is a significant fraction of CDF that would have vessel failure at low pressure, the conditional probability of containment failure due to alpha mode failure would be about a fraction of one percent of the CDF for these plants. This is a significant contribution to the total early failure probability.

In the Indian Point 2 IPE, the assigned values are 4E-5 and 4E-6 for low and high RCS pressures, respectively. These are about two orders of magnitudes less than those used in NUREG-1150. The conditional probability of alpha mode failure obtained in the Indian Point 2 IPE is about 5E-6 of the CDF. This value is closer to the assigned value for high RCS pressure because in the Indian Point 2 IPE RCS depressurization due to hot leg failure is not accounted for in the determination of early failure. The conditional probability of alpha mode failure for Indian Point 2 would be much higher (in a relative sense) if RCS depressurization due to hot leg failure is accounted for in the IPE. This is because of the high likelihood of hot leg failure with the RCS at pressurizer safety valve setpoint pressure and the high probability of PDSs at such RCS pressure (over 50% of all Indian Point 2 PDSs). The small contribution of alpha mode failure for Indian Point 2 is therefore partly attributable to the small probability values assigned in the IPE for alpha mode failure and partly due to the omission of RCS depressurization due to hot leg failure. Because of the significant uncertainty regarding alpha mode failure, the values used in the Indian Point IPE cannot be judged as unreasonable. However, the treatment of alpha mode failure in the Indian Point 2 IPE explains partly the lower conditional probability of early containment failure obtained in the Indian Point 2 IPE as compared with the values found in the NUREG-1150 study.

Containment bypass failure results primarily from SGTR initiated events (86% of bypass CDF). This is followed by induced SGTR (13% of bypass CDF) and ISLOCA (1% of bypass CDF). NUREG-1150 data are used in the Indian Point 2 IPE to determine the probability of induced SGTR. The effect of restarting the RCPs, which is considered in some other IPEs as a mechanism that may increase the potential of induced SGTR, is not addressed in the Indian Point 2 IPE.

The conditional probability of late containment failure is 9% of total CDF for Indian Point 2. This probability value is between those calculated in NUREG-1150 for Surry and Zion. For Indian Point 2, the primary cause of late containment failure is containment overpressurization due to the loss of containment heat removal capability. There is a small contribution from long-term containment failure, occurring more than 24 hours after vessel failure, primarily due to basemat melt-through (1% of CDF).

The C-Matrix, which shows the conditional probabilities of STCs for the PDSs, is not provided in the submittal. However, it can be derived easily from the data presented in Table 4.7-1 which shows the conditional probabilities of PDSs for the STCs or from Figures 4.5-1 through 4.5-31 which show the STCs and the associated conditional probabilities for the CET end states of the 31 PDSs.

2.4.2.4 Impact on Equipment Behavior

The probability of containment spray and fan cooler failure due to adverse conditions is assessed in the IPE. The situations considered in the IPE include: (1) an early containment failure directly causing spray and/or fan cooler failure; (2) excessive debris in the sump causing spray pump failure; (3) a failure of the spray pump motors and/or fan cooler units (FCUs) due to harsh environmental conditions; or (4) excessive debris in the upper containment causing FCU failure. In addition, recirculation spray failure late in the accident due to containment failure and harsh environmental conditions inside the containment and in the Auxiliary Building are also evaluated. These situations

provide reasonably good coverage of the conditions that may cause the failure of containment spray and the FCUs. Although the values used in the IPE depend heavily on engineering judgment, considerable discussion is provided in the IPE submittal on the derivation of these values.

According to the submittal there is a significant degree of diversity and redundancy in the Indian Point 2 containment heat removal systems. Sufficient containment heat removal can be provided by the containment spray system driven by either a recirculation pump, located in the lower compartment of the containment and taking suction from the recirculation sump, or a RHR pump, located outside the containment and taking suction from the containment sump. Sufficient heat removal can also be provided by operation of three of the five fan coolers, located in the annular compartment below the operating decks. Geometrical factors like relative location were considered in assigning the probabilities of loss of function due to harsh environmental conditions.

2.4.2.5 Uncertainties and Sensitivity Analysis

Sensitivity calculations performed in the Indian Point 2 IPE are discussed in Section 4.8 of the submittal. The parameters that are considered for sensitivity evaluation include:

- Induced RCS failure -- hot leg failure and induced SGTR,
- In-vessel debris cooling,
- Early containment failure -- DCH load and containment failure pressure,
- Recirculation spray failure,
- Ex-vessel debris cooling.

The effects of changing the probabilities of alpha mode failure and containment isolation failure, and changing the frequencies of the bypass sequences, are also briefly discussed. For these parameters, sensitivity calculations are not performed and only observations are provided in the submittal. According to the submittal, changing the assigned probability values of alpha mode failure and isolation failure by a factor of ten would not significantly affect the IPE results. This is because of the extremely small probability values used in the base case of the IPE. As for the bypass sequences, only those that involve Type I release are discussed in the sensitivity study. As noted in the submittal, the frequencies of these events are obtained in the Level 1 analysis and they are not affected by the CET analysis.

According to the sensitivity study, containment performance is not significantly affected by the changes of the probability values assigned to the parameters considered in the study (i.e., hot leg failure, the intensity of the DCH load, and containment failure pressure, from 126 psig to 105 psig). For example, sensitivity calculations show that the conditional probability of early containment failure changed from 6.4% (of CDF) for the base case to 7.3% for the case with more intense DCH involving 50% core mass and 6.2% for the case with less intense DCH involving 10% of core mass. This lack of sensitivity is partly due to the grouping of containment bypass with early containment failure in the IPE. Since containment bypass is the dominant contributor to early containment failure, 6.2% of CDF, the sensitivity of early containment failure to DCH uncertainty is not shown clearly in the comparison. For example, if containment bypass is excluded from the early failure category

for the above case, the probability of early containment failure due to DCH then changes from about 0.1% for the base case to about 1% for the case with more intense DCH, and is negligible for the case with less intense DCH. It must be noted, however, that although the containment failure probability is sensitive to the DCH in a relative sense, the absolute value of the containment failure probability due to DCH is small (e.g., less than about 1%).

The probability values assigned in the IPE for induced SGTR is 0.018 for RCS at high-high pressure. Results of the sensitivity study show that the probability of the release category associated with SGTR without stuck-open valves, which induced SGTR is grouped to, is reduced from 0.049 for the base case to 0.04 if induced SGTR is assumed not to occur. The contribution of 0.009 to the conditional probability from ISGTR is consistent with the assigned probability of 0.018 and the fraction of PDSs at high-high pressure (about 50%).

For hot leg failure, the probability values assigned in the IPE are 0.72 for high-high pressure and 0.034 for high pressure. According to the results presented in the licensee's response to the RAI, the probability of in-vessel recovery is significantly reduced if hot leg failure is assumed not to occur (from 31% to 5% of CDF), and moderately increased if hot leg failure is assured (from 31% to 43%). The sensitivity study also shows that the effect of hot leg failure on containment failure is not significant.

In the evaluation of in-vessel debris cooling, the probability of core melt arrest in-vessel is calculated for the various cases with different in-vessel recovery mechanisms. Results show that the mechanism that has the most significant effect is hot leg failure, which reduces the RCS pressure to below the shutoff heads of the injection systems to allow them to provide coolant injection to the RCS. The probability of successful core cooling under this coolant make-up condition is assigned a value of 0.9 in the IPE base case. This results in a 0.31 probability of core melt arrest in-vessel. This probability drops to 0.05 if core cooling is assumed not successful. This is basically the same as that obtained from assuming hot leg failure would not occur. The effects of in-vessel recovery on containment failure are not discussed in the submittal.

The other parameters that are considered in the sensitivity study affect primarily late containment failure probabilities. Calculation results show that the probability of long-term containment failure due to basemat melt-through increases from 0.01 to 0.13 if ex-vessel core debris is assumed not coolable. Since only 25% of the cases with non-coolable ex-vessel core debris are assumed in the IPE to result in long-term failure, the probability of long-term containment failure would increase if this probability value (i.e., 25%) increases. For example, if it is assumed that 50% of the cases with non-coolable ex-vessel core debris would result in basemat melt-through, then the probability of long-term containment failure would increase from 0.02 to 0.26, instead of from 0.01 to 0.13. Although the long-term failure probability seems to be sensitive to some of the parameter values assumed in the IPE, it is not a significant concern. This is because of the significant uncertainties on these parameters and the long time it takes to melt-through the basemat.

The sensitivity study also shows that the probability of late containment failure increases significantly (from 8 to 53% of CDF) if containment spray is assumed to always fail. In Indian Point

2, the containment spray system can be supplied by either the recirculation pumps, located in the containment, or the RHR pumps, located outside the containment. These pumps also take suction from different sumps in the containment. The probability of losing the containment spray system would appear to be small. However, this does indicate the importance of the containment spray system on containment performance. Special attention should therefore be paid to assure its operation during a severe accident.

The sensitivity study provided in the Indian Point 2 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 and CPI

2.5.1 Evaluation of Decay Heat Removal

2.5.1.1 Examination of DHR

The IPE addresses decay heat removal (DHR). Several methods of DHR are mentioned, including secondary cooldown and depressurization (using either AFW or recovery of main feedwater), feed and bleed, safety injection, and recirculation cooling. CDF contributions were estimated for each of the following DHR methods: auxiliary feedwater cooling (48% contribution to the CDF), feed and bleed (32%), low pressure recirculation cooling (18%) and high pressure recirculation cooling (6%).

Failure of the AFW system was determined to be a major contributor to the total CDF. However, the licensee states that in absolute terms, the CDF due to loss of this system is similar to the contribution in other PRAs (comparison is made with the Seabrook and Surry IPEs). The contribution due to failure of feed and bleed is strongly coupled to that of auxiliary feedwater (i.e. feed and bleed is attempted after the AFW has failed). With respect to recirculation decay heat removal, again the case is made that the absolute contribution is similar to that found at Seabrook and Surry.

Major contributors to the failure of auxiliary feedwater and feed and bleed are identified and their contribution quantified. For the auxiliary feedwater, hardware failures of the pumps dominate, contributing over 80% to the AFW failure. Common cause failure of the two motor driven pumps in conjunction with a random failure of the turbine driven pump accounts for 55% of the AFW failures. In contrast, operator errors are important in failure of feed and bleed, contributing 40% to this failure. Another 42% is due to failure of the PORVs to open, and the rest (18%) is due to failure of the block valves to open.

The operating philosophy of IP-2 with respect to operation with block valves closed had changed prior to the IPE, from running with the block valves closed to running with the block valves open whenever possible. The valves are only closed when PORV leakage is experienced. The licensee estimates that each block valve is now closed about 25% of the time. In addition a design change to the PORVs was implemented during the 1995 refueling outage (not credited in the IPE) which

is expected to eliminate the cause of past leakage. The licensee is monitoring the PORV performance (RAI Responses).

As operator error is the major cause of feed and bleed failure, the licensee is considering changing the EOPs to allow earlier entry into the feed and bleed procedure. This procedure is initiated following loss of secondary side cooling. Possible disadvantages to this potential improvement are noted by the licensee in the RAI Responses. It is recognized that this is a severe action and is executed only after attempts to recover auxiliary feedwater or reestablish main feedwater have failed. These recovery actions were not modeled in the IPE, and thus the submittal argues that the potential for core damage following loss of auxiliary feedwater is overstated. An earlier cue for the operator to establish feed and bleed would detract from these recovery actions, thus reducing chances of successfully recovering feedwater.

The licensee states that due to the significance of this scenario (i.e. establishing feed and bleed following loss of feedwater) particular attention is paid to it in operator training.

2.5.1.2 Diverse Means of DHR

The IPE evaluated the diverse means for DHR, including: use of the power conversion system, feed and bleed, auxiliary feedwater, and ECCS. Depressurization using the secondary system was considered for small LOCA accidents when the HHSI was unavailable. Cooling for RCP seals was considered. In addition, containment cooling was addressed.

2.5.1.3 Unique Features of DHR

The unique features at Indian Point 2 that directly impact the ability to provide DHR are described in Section 1.2 ("Kev Features").

2.5.2 Other GSIs/USIs Addressed in the Submittal

No GSIs or USIs, other than USI A-45 (DHR Evaluation) are addressed in the submittal.

2.5.3 Response to CPI Program Recommendations

The CPI recommendation for PWRs with a dry containment is to evaluate containment and equipment for vulnerabilities to localized hydrogen combustion and the need for improvements. A level 2 plant walkdown was performed by Indian Point 2 IPE PRA personnel. This was augmented by examination of a video laser disc walkdown record that was available for areas not considered accessible because of high radioactivity or other reasons. No areas were found in the containment where significant pockets of hydrogen could form and the different volumes in the containment were found to be well connected. Local detonation was postulated not to occur due to rapid mixing that would ensue within containment volumes.

The effect of harsh environmental conditions on equipment survivability is assessed in some detail in CET quantification. Although the quantification involves significant engineering judgment, the assessment is logical and reasonable. Additionally, there is a significantly degree of diversity and redundancy in the Indian Point 2 containment heat removal systems, and they are geometrically separated inside and outside of the containment.

2.6 Vulnerabilities and Plant Improvements

2.6.1 Vulnerability

According to the IPE submittal, vulnerabilities are defined based on the application of the NUMARC Severe Accident Closure Guidelines [NUMARC 91-04], and the results of various sensitivity studies. The licensee used NUMARC screening criteria on functional sequence groups to determine if actions are needed to improve the plant risk profile. The NUMARC criteria define various types of plant improvements which may be considered, depending on the absolute CDF, or fractional contribution to CDF, of a functional group of sequences.

For most functional sequences in the IPE, the conclusion is that no further action is needed. Only functional sequence group 1A would require additional attention with regard to exploring further mitigative measures. This group is derived from the General Transient with failure of auxiliary feedwater and failure of primary bleed.

The licensee notes certain conservatisms in the modeling of these sequences (e.g. turbine driven AFW pump mission time, no credit given to recovery of main feedwater/condensate, assuming that the PORV block valves are normally closed which is no longer the operating practice). The difficulty of executing primary bleed is recognized as resulting from the relatively short time window (450 seconds). The licensee notes that consideration will be given to cueing operators earlier to start the feed and bleed operation.

Therefore no vulnerabilities were explicitly called out. No attempt was made to define a vulnerability based on the contribution to CDF of any single accident sequence, or on the importance ranking of any single system or component.

No containment vulnerabilities were identified as a result of performing the IPE.

2.6.2 Proposed Improvements and Modifications

The IPE took credit only for plant modifications and improvements that were complete at the time of the freeze date. The following potential improvements have been identified in the IPE and their status noted in the IPE and the RAI responses:

1) Gas Turbine No. 2 (which is the most reliable of the three gas turbines) has been equipped with full blackstart capability. This provides an additional means of recovering power following station blackout sequences.

2) A sixth fan has been installed in the EDG building and the power supply configuration to the EDG building fans has been improved to eliminate the earlier dependency between failures of EDGs 21 and 22 (which had been used to power all the fans in the EDG Building) and EDG 23 (which previously had no such capability). Now each EDG powers two of the fans, with one fan per EDG being sufficient to provide ventilation. This change is expected to reduce the CDF by about 5%.

3) Periodic testing of all EDG Building fans has been instituted.

4) The operating position of the PORV Block Valves is being tracked to provide a fuller basis for modeling. Operating and technical personnel have been made aware of the risk importance of these valves. In addition, PORVs with a more leak resistant design have been installed during the 1995 refueling outage.

5) Changing the EOPs to enable an earlier initiation of feed and bleed is being considered. There are possible downside effects from this action in that this may detract from operator efforts to recover the more benign method of core cooling, i.e. secondary cooling by recovering either the auxiliary feedwater or the main feedwater/condensate system. Therefore it is not clear that this modification will be implemented.

No quantitative impact of these changes on the CDF is available at this time but the licensee intends to incorporate modeling of these modifications into the next PSA update (RAI responses).

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

Strengths of the IPE are as follows: Plant specific data were used where possible to support the quantification of initiating events and component unavailabilities.

No major weaknesses of the IPE were identified. Some common cause MGL factors seem low compared to NUREG/CR-4550 data, however the licensee seems to have done a thorough analysis of common cause data as it applies to their plant, along with a cause-defense analysis.

The IPE determined that failures in the auxiliary feedwater system (dominated by hardware pump failures) and in the primary feed and bleed operation (dominated by operator failures and failure of PORVs to open) are the principal contributors to core damage.

LOCAs are a relatively high contributor to the CDF (33%) due to operator error in aligning recirculation.

The contribution of station blackout (14%) is due to some peculiar plant features. On the one hand, electric power recovery is enhanced by the fact that two of the three gas turbines have blackstart capability (at the time of the IPE). On the other hand distribution of power from the three EDGs to the safeguards equipment is staggered (i.e. not every EDG will power every system, as the number of EDGs and number of safety equipment trains is not always matched). For example, EDG 21 does not power an auxiliary feedwater pump, as there are two such motor driven pumps, powered from the other two EDGs. While the hardware (i.e., breakers) exists for cross-connecting emergency buses, doing so is against the technical specifications and is under strict administrative control, and is not credited in the IPE (RAI Responses). Another detrimental feature of the emergency power system is the fact that a start of EDG 23 only will not constitute success of emergency power because of the arrangement of the power supply to the EDG Building fans (i.e., only EDG 21 and EDG 22 are connected to the EDG Building fans). The resistance of the plant to station blackout has been improved (since the IPE) by adding the blackstart capability to the third gas turbine and by correcting alignment of the EDG-to-EDG Building fan power supply.

As was noted previously, several improvements have been completed as a result of insights from the IPE. The CDF impact of these improvements is not known.

The HRA review of the Indian Point 2 IPE submittal did not identify any significant problems or errors. A viable approach was used in performing the HRA and although there were some minor potential weaknesses, 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 indicates that utility personnel were involved in the HRA and that the walkdowns, documentation reviews and simulator observations represented a viable process for confirming that the HRA portions of the IPE represent the as-built-as operated plant.

2) The submittal indicated that the analysis of pre-initiator actions included both miscalibrations and restoration faults. Eleven restoration events and their HEPs were listed in the submittal. In response to an NRC RAI, treatment of miscalibration events was discussed in more detail than in the submittal, but exactly how miscalibration HEPs were determined was not clearly explained. In addition, a listing of all of the miscalibration events was not provided and it was unclear exactly how many were actually modeled. Treatment of miscalibration events could be a minor weakness of the pre-initiator analysis.

3) The Indian Point 2 IPE did not make an explicit distinction between response and recovery type post-initiator human actions. However, the submittal states that no credit was taken for human actions that were not supported by written operating procedures and that were not demonstrated to be viable. A screening analysis of the post-initiator events was not conducted. All events received detailed quantification and dependencies and context effects were considered. The quantification method and its application was acceptable and the overall analysis appeared reasonably thorough. The only potential weaknesses in regards to the post-initiator analysis concerned the extension of the basic methodology to better amount for PSFs and dependencies. While the "extensions" seemed to improve the methodology, evidence for the validity of the assumptions underlying the improvements were not provided. Nevertheless, the extensions did have significant face validity, i.e., they appeared reasonable.

4) Manual backups to failed automatic actions were conservatively not modeled. The IPE submittal noted that this facilitates consideration of dependencies, which is true. However, by not taking credit for manual recoveries of failed automatic actions, the results of the IPE could be somewhat distorted. To assume that such recoveries would fail may be conservative, but is not realistic. At a minimum, such an approach prevents the licensee from getting the full benefits of the IPE and may impact the discovery of issues related to accident management.

5) The licensee did not identify important human actions through the use of importance measures in the submittal. The submittal did provide a sufficient discussion of operator actions in dominant sequences and a sensitivity analysis for human action events in truncated sequences was performed and discussed. Thus, information regarding important human actions was provided.

The IPE uses a small containment event tree (CET) with 8 top events for back-end analysis. Decomposition event trees are used to quantify the CET top events. The CET quantification relies heavily on NUREG-1150 data, and plant specific MAAP calculations are used to assist CET quantification and define source terms. The interface between the front-end and back-end analyses is accomplished by the development of a set of PDSs, which are defined by the front-end core damage sequences and the status of the containment systems. The PDS definition is

adequate. The CET is well structured and easy to understand, and the quantification process of the CET is systematic and traceable.

The important points of the technical evaluation of the Indian Point 2 IPE back-end analysis are summarized below:

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

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

- The quantification of the CET relies heavily on NUREG-1150 data. In general, similarity with the NUREG-1150 plants is established in the IPE and the use of the data seems justifiable. Engineering judgment of the analyst is used in the quantification of some parameters. This usually involve the use of a logical structure and the assignment of qualitative probability values for the parameters used in the logic structure. The approach and the assigned values seem reasonable. Parameters with significant uncertainties are included in the sensitivity study.

- While the effect of RCS depressurization due to hot leg creep rupture is considered in the CET, its effect on early containment failure is not included in the early failure DET. Although the contributions from the mechanisms that are affected by RCS depressurization are not significant in the Indian Point 2 IPE, such that this omission is not expected to have a significant effect on the containment failure profile for Indian Point 2, it does represent an inconsistency in the CET modeling.

- Accident sequences are selected in the IPE for MAAP calculations to provide data to assist CET quantification and for estimating the source terms. However, the selection criteria is not discussed in the IPE submittal. The relationship between the selected sequences and the accident sequences binned to the PDSs or the source term categories is not established or discussed in the submittal. Nonetheless, the sequences selected for MAAP calculation seem to provide a reasonable representation of the source term categories.

- Except for the recovery of ac power for the SBO sequences, operator recovery actions, such as those for RCS depressurization and recovery of containment heat removal systems, are not included in the IPE. Although this would appear to provide a conservative estimate of accident progression results, it limits the use of the CET structure to investigate the likely benefit of these recovery actions.

- The licensee has addressed the recommendations of the CPI program.

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