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Clinton Power Station

Technical Evaluation Report on the Individual Plant Examination Front End Analysis

NRC-04-91-066, Task 42

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Prepared for the
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This report summarizes the results of our review of the front-end portion of the Individual Plant Examination (IPE) for the Clinton Power Station. This review is based on information contained in the IPE submittal [IPE Submittal] along with the licensee's responses [RAI Responses] to a request for additional information (RAI)[1].

E.1 Plant Characterization

The Clinton Power Station (CPS) consists of a single unit boiling water reactor (BWR)-6 with a Mark III containment. Clinton began commercial operation in April 1987.

Design features at Clinton that impact the core damage frequency (CDF) relative to other BWR 6 plants are as follows:

- Four hour battery lifetime. With credit for load shedding, the battery lifetime can be extended to 4 hours. However, a 4 hour battery lifetime is less than battery lifetimes at some other BWRs. The 4 hour battery lifetime at Clinton tends to increase the CDF compared to those BWRs with longer battery lifetimes.

- Ability of emergency core cooling system (ECCS) pumps to operate with a saturated suppression pool. The high pressure core spray (HPCS), low pressure core spray (LPCS), and residual heat removal (RHR) pumps can operate with a saturated suppression pool and thus provide core cooling in the event containment cooling is lost. This design feature tends to decrease the CDF.

- Ability to cross-connect the fire protection system for core injection. The fire protection system can be aligned as a source of core injection. The fire protection pumps are diesel-driven. This design feature tends to decrease the CDF. The analysis took credit for this cooling method, though apparently not for station blackout sequences. This cooling method would be of minimal value in station blackout sequences because the automatic depressurization system (ADS) safety relief valves (SRVs) will likely reclose after battery depletion, with a consequential rise in reactor pressure that would make fire protection injection unavailable.

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[1] In responding to the RAI, the licensee states that several updates have been made to the original IPE analysis described in the submittal. Because no details are available for the latest IPE revision other than a total CDF exclusive of flooding, our review is focused on the IPE presented in the submittal. [pp. 2, 5 of RAI Responses]

E.2 Licensee's IPE Process

The licensee developed a Level 2 probabilistic risk assessment (PRA) in response to the requests of Generic Letter 88-20. The freeze date of the analysis was December 31, 1991. It appears that the only exception to the freeze date was the implementation of several station blackout procedures that were credited in the analysis.

The licensee had the primary role in each phase of the IPE, including: overall project management, reviews of interim analysis products, and critical analysis and evaluation of all results. Consultants used in the analysis were from Tenera, L. P., Fauske and Associates, and Westinghouse.

Major documentation used in the IPE included: piping and electrical diagrams, operating and emergency procedures, vendor manuals, system descriptions, maintenance work requests, surveillance logs, Technical Specifications, and licensee event reports (LERs). Plant walkdowns were also conducted to support the IPE.

An independent review of the IPE analysis was performed. The IPE independent review team (IIRT) consisted of six members of the utility staff. The chairman of the IIRT is the director of nuclear safety. Four of the other members have Clinton SRO licenses, while the remaining individual has broad maintenance experience.

The licensee intends to maintain the PRA as a living document to support future plant operations.

E.3 Front-End Analysis

The methodology chosen for the Clinton IPE front-end analysis was a Level 1 PRA. The small event tree/large fault tree technique with fault tree linking was used. Accident sequence quantification was performed with the Cut Set and Fault Tree Analysis (CAFTA) and Set Equation Transformation System (SETS) codes.

Event trees were developed for all classes of initiating events. Support systems were modeled with fault trees and linked with the appropriate frontline system fault trees. An importance analysis was performed and described in the submittal. Several sensitivity analyses were performed on the front-end results.

The success criteria were based on Modular Accident Analysis Program (MAAP) calculations. Core damage is defined as a reactor level less than two-thirds the length of the active fuel for more than 4 minutes or MAAP results with a fuel temperature of 2,200 deg. F or more.

The IPE quantified 12 initiating events exclusive of internal flooding: 5 loss of coolant accidents (LOCAs); 4 generic transients, including loss of offsite power (LOSP); and 3 special initiating events representing loss of support systems. The number of initiating events considered in the flooding analysis was not specified.

Plant-specific data were used for test and maintenance unavailabilities. However, component unavailabilities due to failures were entirely based on generic data, with the possible exception of diesel generator start failures. All the initiating events were based on generic data, though some plant-specific considerations were included in the development of the LOSP initiating event frequency.

The Multiple Greek Letter (MGL) method was used to model common cause failures. The source of MGL data was not specified.

The total point estimate CDF for Clinton is 2.6E-05/yr[2], including internal flooding. The CDF contribution from flooding is 1.6E-06/yr[3].

The initiating events that contribute most to the CDF and their percent contribution are listed below[4]:

Loss of off-site power	46%
Transient w/o isolation from main cond.	18%
Transient with isolation from main cond.	16%
Loss of DC bus	5%
Inadvertent Open Relief Valve (IORV)	4%
Loss of Feedwater	4%

Core damage contributions by accident type are listed below:

Transients	52%
Station blackout	37%
Internal Flooding	6%
LOCA (includes IORV)	4%
Anticipated transient without scram (ATWS)	0.5%
Interfacing Systems LOCA (ISLOCA)	negligible

The most important non-initiating events are (in order):

- Failure to recover off-site power in 0.5 hours
- Independent sub-tree containing HPCS failure basic events

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[2] The most recent update of the IPE predicts a CDF exclusive of flooding of 5.5E-06/yr. [p. 5 of RAI Responses]

[3] As used here and in other portions of this report, the term "yr" refers to a reactor year.

[4] A more complete set of initiating event CDF contributors is provided in Table 2-5 of this report.

- Basic event representing recovery of HPCS failures
- Independent sub-tree containing reactor core isolation cooling (RCIC) failure basic events
- Basic event representing recovery of RCIC failures
- Operator fails to manually initiate ADS

The Level 1 core damage end states were binned into accident classes to form the beginning states for containment event trees. This binning process appears to be comparable with similar methods used in other PRA/IPE studies.

E.4 Generic Issues

The decay heat removal (DHR) contribution to CDF was derived by eliminating from accident sequence cutsets failures of systems that cannot remove decay heat. Systems not able to remove decay heat include HPCS, RCIC, LPCS, ADS, and fire protection. The CDF due to loss of DHR was estimated to be 5.2E-06/yr. The licensee states that this DHR-related CDF estimate is pessimistic, as additional methods of DHR were not credited, for example RHR lined up to the fuel pool cooling and cleanup system.

As pointed out by the licensee, the unresolved safety issue (USI) A-45 study recommends that DHR-related CDF contributions should not be greater than 1E-05/yr. The Clinton DHR-related CDF was determined to be 5.2E-06/yr. No DHR-related vulnerabilities were identified.

The licensee does not address any generic safety issues(GSIs)/USIs other than DHR. The licensee states that there are no open generic issues at Clinton.

E.5 Vulnerabilities and Plant Improvements

The licensee selected the following definition of a plant specific vulnerability:

- New or unusual means by which core damage or containment failure occur as compared to those identified in other PRAs, or
- Results that suggest the plant CDF would not be able to meet the NRC's safety goal for core damage (1E-04/yr), or
- Systems, components, or operator actions that control the core damage result (i.e., greater than 90%).

Based on the above criteria, the licensee determined that there are no vulnerabilities at Clinton.

The following front-end plant improvements were identified as a result of the IPE:

- Operator training to emphasize importance of maintaining off-site power

- Operator training to emphasize importance of manual ADS initiation
- Modify HPCS surveillance procedure to test suppression pool suction path
- Install bypass line to allow easier use of fire protection system for vessel makeup

None of these improvements was credited in the IPE version reported in the submittal. The last two improvements (HPCS surveillance and fire protection bypass line) would each reduce the IPE CDF by about 13%. Estimates of CDF reductions for the other two improvements were not available.

E.6 Observations

Because Clinton began commercial operation in April 1987, there is a relatively limited operational history from which to derive plant-specific failure rates. While plant-specific data were used in the IPE for test and maintenance unavailabilities, component hardware failures were entirely based on generic data (with the possible exception of diesel generator start failures). Initiating event frequencies were for the most part also based on generic data. To support this wide use of generic data, the licensee cited instances where plant-specific failure data are comparable to or better than corresponding generic data.

Some other plants with limited operational experience have used plant-specific data to update generic data via a Bayesian process. It is not clear why the Clinton IPE did not use a similar approach. In our judgment, the limited use of plant-specific data represents a weakness of the Clinton analysis. While it might be argued that the use of generic data provides an upper bound to the total CDF, the relative CDF contributions of various sequences and failure events may be distorted.

It is also noteworthy that the Clinton IPE credited local repair of various equipment items and systems, including diesel generators, pumps, valves, and instrumentation. It is positive that the licensee has attempted to credit a variety of repair activities to reflect the operation of the as-built, as-operated plant. However, IPE/PRA studies typically limit credit for local equipment repair activities to diesel generators, as there is comparatively more experience for repair of diesel generators than for other components and systems. It is further noted that the Clinton IPE has taken credit for up to two component/system repair actions per accident sequence cut set. Credit for multiple repair activities within a given cut set is also not typically done in IPE/PRA studies. The licensee states that the credited repair activities included in the Clinton IPE are based on a demonstrated capability of the plant to field multiple repair teams during actual emergency exercises. However, the quantification of repair activities is based on a generic EPRI database, and it is not clear how accurately the generic EPRI data would reflect the Clinton plant during an actual accident condition given the uncertainties inherent in predicting human actions. It is also noted that the IPE data for non-recovery of common cause diesel generator failures are one to two orders of magnitude lower (more optimistic) than industry experience used in the Accident

Sequence Evaluation Program (ASEP) as reported in NUREG/CR-4550 (Rev. 1, Methodology).

As part of a response to an NRC Staff request for clarification of equipment repair models, the licensee performed a sensitivity analysis that involved removal of all credited equipment repair recoveries, except those involving off-site power, DC power, and operator actions done from the control room. With this model change, the baseline CDF (exclusive of internal flooding[5]) increased by a factor of 1.44 (from 2.49E-05/yr to 3.57E-05/yr). The relative contributions of individual accident sequences were not significantly altered as a result of this sensitivity study. In no instances were increases in individual accident sequence frequencies greater than 2.4. While two new sequences were introduced as a result of the sensitivity analysis, their frequencies were less than 1E-08/yr.

While the Clinton IPE equipment repair model is more optimistic that repair models typically used in other IPE/PRA studies, the licensee's sensitivity analysis demonstrates that this repair model has not significantly affected the CDF or accident sequence profile. Therefore, it is our judgment that the licensee's equipment repair model does not represent a weakness of the IPE.

Significant level-one IPE findings are as follows:

- Operator failure to manually initiate ADS for use of low pressure injection contributes about 24% to the total CDF. The licensee does not consider failure of operators to depressurize as a vulnerability because this action has been emphasized in training and has been judged unlikely to induce operator error.

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[5] No credit was given for flood-related operator mitigating actions.

1. INTRODUCTION

1.1 Review Process

This report summarizes the results of our review of the front-end portion of the IPE for Clinton. This review is based on information contained in the IPE submittal [IPE Submittal] along with the licensee's responses [RAI Responses] to a request for additional information (RAI)[6].

1.2 Plant Characterization

The Clinton Power Station (CPS) consists of a single unit BWR-6 with a Mark III containment. Clinton began commercial operation in April 1987, and has power ratings of 2,894 MWt and 933 net MWe. The Clinton site is located in east-central Illinois. Condenser cooling and the ultimate heat sink for ECCS is provided by Lake Clinton. Sargent & Lundy served as the Architect-Engineer and design consultant for this plant. The River Bend, Perry, and Grand Gulf plants are similar in design to Clinton. [pp. 1.1-1, 1.1-2, 1.3-1, 1.4-2 of UFSAR, 2-9 of submittal]

Design features at Clinton that impact the core damage frequency (CDF) relative to other PWRs are as follows: [pp. 3-84, of submittal, 6.3-5, 6.3-12, 6.3-17 of the UFSAR]

- Four hour battery lifetime. With credit for load shedding, the battery lifetime can be extended to 4 hours. However, a 4 hour battery lifetime is less than battery lifetimes at some other BWRs. The 4 hour battery lifetime at Clinton tends to increase the CDF compared to those BWRs with longer battery lifetimes.

- Ability of emergency core cooling system (ECCS) pumps to operate with a saturated suppression pool. The high pressure core spray (HPCS), low pressure core spray (LPCS), and residual heat removal (RHR) pumps can operate with a saturated suppression pool and thus provide core cooling in the event containment cooling is lost. This design feature tends to decrease the CDF.

- Ability to cross-connect the fire protection system for core injection. The fire protection system can be aligned as a source of core injection. The fire protection pumps are diesel-driven. This design feature tends to decrease the CDF. The analysis took credit for this cooling method, though apparently not for station blackout sequences. The licensee notes that this cooling method

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would be of minimal value in station blackout sequences because the ADS SRVs will likely reclose after battery depletion, with a consequential rise in reactor pressure that would make fire protection injection unavailable. [pp.10, 11 of RAI Responses, pp. 3-32, 3-89, 6-11, 6-12 of submittal]

2. TECHNICAL REVIEW

2.1 Licensee's IPE Process

We reviewed the process used by the licensee 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 submittal appears to be complete with respect to the type of information requested by Generic Letter 88-20 and NUREG 1335. No omissions were noted. [pp. 2-2 of submittal]

The front-end portion of the IPE is a Level 1 PRA. The specific technique used for the Level 1 PRA was the small event tree/large fault tree technique with fault tree linking. [pp. 1-5 to 1-7 of submittal]

Intersystem dependencies were discussed and tables of system dependencies were provided. Data for quantification of the models were provided, including common cause events and human recovery actions. An importance analysis was performed and is described in the submittal. Several sensitivity analyses were performed on the front-end analysis results.

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

The Clinton plant is a single unit site; therefore, multi-unit considerations do not apply to this plant.

The IPE was based on a variety of plant-specific information, including piping and electrical diagrams (P&IDs), operating and emergency procedures, vendor manuals, system descriptions, maintenance work requests, surveillance logs, and Technical Specifications. Plant walkdowns were also conducted to support the IPE analysis. A flooding walkdown was performed to determine flooding sources and potential effects of flooding, including ISLOCA effects. Other walkdowns were made, for example to address HRA considerations and to answer specific questions as they arose during the analysis. [p. 1-4, 2-12, 2-14 of submittal]

The IPE made very limited use of plant-specific failure data. While plant-specific data were used for test and maintenance unavailabilities, component hardware failures were entirely based on generic data (with the possible exception of diesel generator start failures). The initiating events were generally based on generic data, though some plant-specific considerations were included in the development of the LOSP initiating event frequency. In our judgment, the limited use of plant-specific data represents a weakness of the IPE.

The freeze date of the analysis was December 31, 1991. It appears that the only exception to the freeze date was the implementation of several station blackout procedures that were credited in the analysis. These station blackout procedures are described more thoroughly in Subsection 2.7.3 of this report. [pp. 3, 9 of RAI Responses]

The licensee intends to maintain the PRA as a living document to support future plant operations. [cover letter, p. 1-5 of submittal]

2.1.3 Licensee Participation and Peer Review.

The licensee had the primary role in each phase of the IPE, including: overall project management, reviews of interim analysis products, and critical analysis and evaluation of all results. All of the major work tasks were performed by licensee personnel. Consultants used in the analysis were from Tenera, L. P., Fauske and Associates, and Westinghouse. The consultants provided support in several areas, including expertise in specific aspects of PRA and technical review of program products. Technology transfer from the consultants to the licensee's employees was considered as a very important part of the IPE program. [pp. 5-1 to 5-3 of submittal]

Plant system engineers were involved in the IPE process to answer questions related to design, capability, and function of the modeled systems, as well as to review the system models. The system engineers were trained in PRA terminology and methods to support the IPE analysis. A senior management review team (SMRT) consisting of upper level utility management staff was used to provide program oversight and to review the IPE progress and results. [pp. 5-4, 5-6 of submittal]

An independent review of the IPE analysis was also performed. The IPE independent review team (IIRT) consisted of six members of the utility staff, specifically the director of nuclear safety, four individuals with senior reactor operator (SRO) licenses, and one individual with broad maintenance experience. The IIRT members were provided with training related to the PRA process. The licensee has provided a sampling of IIRT comments. [pp. 23 to 25 of RAI Responses, pp. 5-5, 5-6 of submittal]

2.2 Accident Sequence Delineation and System Analysis

This section of the report documents our review of both the accident sequence delineation and the evaluation of system performance and system dependencies provided in the submittal.

2.2.1 Initiating Events.

The specific categories of initiating events utilized in the IPE are listed below: [pp. 3-2 to 3-4, 3-6, 3-37 to 3-39 of submittal].

Transients:
- Loss of offsite power (LOSP)
- Loss of feedwater
- Transient with isolation from main condenser
- Transient without isolation from main condenser
- Inadvertent/stuck open safety relief valve (IORV)

Special Initiators:
- Loss of instrument air
- Loss of service water
- Loss of non-safety DC bus

LOCAs:
- Small (within capacity of RCIC system)
- Medium (beyond capacity of RCIC system)
- Large (sufficient depressurization to allow use of low pressure injection systems)
- ISLOCA (7 separate categories)

Internal Flooding:
- Number of initiating events not provided

Failures of individual AC buses were excluded as initiating events. The licensee acknowledges that loss of a single safety-related AC bus could cause a transient with isolation due to closure of the main steam line isolation valves (MSIVs). However, loss of a safety-related AC bus was omitted from the analysis because its expected frequency (8.7E-04/yr) is about 3 orders of magnitude lower than the frequency for a transient without isolation 1.7/yr. In our judgment, this rationale for omitting safety-bus AC bus failures is not necessarily supportable. While the expected frequency for the AC safety-bus loss is 3 orders of magnitude lower than the frequency for transient without isolation, the AC bus loss represents an automatic failure of the safety equipment powered by that bus that otherwise might have been available to mitigate the transient. In contrast, the logic model for "transient without isolation" does not include this automatic failure. [p. 3 of RAI Responses]

The failure of an individual non-safety AC bus would lead to essentially the same conditions as a loss of service water. The loss of a non-safety AC bus was omitted from the analysis because its expected frequency is about an order of magnitude lower than the frequency for loss of service water (1.75E-03/yr). [pp. 3, 4 of RAI Responses, p. 3-38 of submittal]

Loss of an individual non-safety DC bus was modeled as an initiating event, as it will cause a plant trip. In contrast, the loss of a safety-related DC bus would not cause a plant trip, and therefore was not modeled as an initiator. Total loss of DC was not modeled as an initiating event, because failure of all 6 independent battery-charger subsystems (4 safety-related, 2 balance-of-plant) was not judged to be credible. [p. 4 of RAI Responses, p. 3-84 of submittal]

HVAC failures are encompassed in the quantification of the transient initiating events, instead of being modeled separately. For example, control room HVAC failures are included in the quantification of "transient with isolation." Because of system redundancies, the contribution of control room HVAC failure to this transient initiating event frequency is stated to be very small. In our judgment, the licensee's rationale for grouping loss of HVAC events into the transient initiating events is not necessarily supportable. A loss of HVAC to a plant area may disable certain mitigating system equipment (beyond that required to cause a plant trip) that otherwise might have been available to mitigate the transient. It does not appear that the logic models for transient events include the possibility of these types of consequential failures. [p. 4 of RAI Responses]

The IPE does not have separate initiating events for loss of component cooling water (CCW) or turbine building closed cooling water (TBCCW). The CCW system provides cooling for the service air compressors and recirculation pump seal coolers (though the shutdown service water system can also provide backup cooling to the recirculation pump seals). Loss of cooling to the service air compressors will lead to loss of instrument air, while loss of recirculation pump seal cooling can result in a pump seal LOCA. The TBCCW system provides cooling for major non-safety components in the turbine building, including the condensate booster pumps. Presumably, loss of CCW and TBCCW have been included in the quantification of other initiating events. [pp. 3-38, 3-86, 3-88, 3-157 of submittal]

Seven separate categories of ISLOCA were addressed in the analysis. These seven categories are: (1) LPCI injection lines, (2) LPCS injection line, (3) shutdown cooling suction line, (4) RPV head spray line (from RCIC and LPCI loop B), (5) HPCS line, (6) feedwater lines, and (7) shutdown cooling return lines. Lines eliminated from the ISLOCA analysis included lines with a diameter less than 1.5 inches, and CRD injection lines. [pp. 3-11, 3-12, 3-39 of submittal]

Initiating event frequencies for LOCAs were based on WASH-1400. The ISLOCA events were quantified by using the methods described in NUREG/CR-5124, supplemented by input from WASH-1400, the IDCOR BWR IPE Methodology, EPRI pipe failure data (no reference provided), and the GESSAR PRA. [pp. 3-11, 3-12, 3-40 of submittal]

The frequency for LOSP was derived from industry data and plant-specific site location data with methods described in NUREG 1032 and NUMARC 87 00. Frequencies for other generic types of transient initiating events were based on data from the NUREG/CR-4550 Grand Gulf study. Plant-specific data were not used for generic transients due to limited plant operating experience. However, the licensee made a comparison of limited plant data and NUREG/CR-4550 Grand Gulf data for several transient categories. This comparison shows good agreement between the limited plant-specific data and NUREG/CR-4550 Grand Gulf data. [pp. 3-12 to 3-14, 3-40 of submittal]

Initiating event frequencies for the loss of plant service water and instrument air were based on system logic models. The initiating event frequency for loss of a non-safety DC bus was based on data from NUREG-0666. [pp. 3-10, 3-12 to 3-15 of submittal]

Frequencies for equipment failures associated with flooding initiating events were extracted from WASH-1400, PRAs for Seabrook and Oconee, and NUREG/CR 1363. [pp. 3-188 to 3-190, 3-211 of submittal]

Except as discussed below, the initiating event frequencies are comparable to those typically used in other BWR IPE/PRA studies.

The Clinton IPE used a frequency of 1E-03/yr for small LOCAs. However, there appear to have been 2 instances of recirculation pump seal failures during the Clinton plant history. Given that the Clinton IPE does not have a separate initiating event for recirculation pump seal LOCAs, it appears that the small LOCA frequency has been underestimated. In other typical BWR PRA/IPE studies, the frequency of recirculation pump seal LOCAs is approximately 1E-02/yr. [p. 2 of RAI Responses, p. 3-37 of submittal]

2.2.2 Event Trees.

The following event trees were used in the analysis: [pp. 3-22 to 3-62 of submittal]

Transient with isolation from main condenser
Transient without isolation from main condenser
Loss of feedwater
Inadvertent/stuck open safety relief valve (IORV)
Loss of offsite power (LOSP)
Station blackout
Loss of instrument air
Loss of service water
Loss of non-safety DC bus
Small LOCA
Medium LOCA
Large LOCA
ISLOCA
ATWS

The front-end portion of the analysis was based on a 24 hour mission time, while the back-end analysis assumed a 48 hour mission time. [pp. 3-16, 3-164, 4-22 of submittal]

Success criteria used in the analysis were based on Modular Accident Analysis Program (MAAP) calculations. Core damage was defined as a reactor level less than two-thirds the length of the active fuel for more than 4 minutes or MAAP results with a fuel temperature of 2,200 deg. F or more. Decay heat levels typical of conditions immediately following reactor trip were used in these calculations, with no credit for spray or steam cooling. [pp. 1-7, 3-30, 3-212 of submittal]

The IPE assumes that the LPCS, the HPCS, and the RHR pumps (in the LPCI mode) do not lose suction after loss of containment heat removal or containment depressurization following containment venting or containment failure unless the failure is in the suppression pool. Per design, sufficient NPSH is expected to remain available to operate these pumps with the suppression pool at saturation conditions. [p. 60 of RAI Responses, pp. 6.3-5, 6.3-12, 6.3-17 of UFSAR, 3-28 of the submittal]

Clinton has a suppression pool makeup system, which is designed to dump water from an upper pool down into the suppression pool during post-LOCA conditions. This purpose of this added water is to ensure that adequate water exists in the suppression pool given that inventory during recirculation is diverted out the break from the suppression pool into the drywell. However, the licensee states that upper pool dump is not required for maintaining adequate NPSH for the ECCS pumps in the event of a LOCA. [pp. 3-29, 3-76 of submittal]

As long as the reactor is shutdown and core damage is averted via ECCS cooling, loss of containment cooling will not cause containment failure during the 24 hour front-end mission time. The licensee states that containment cooling is not required due to the relatively large suppression pool volume and free air volume. Because ECCS systems would not be affected during the front-end mission time, containment venting was not required or modeled in the front-end analysis. [p. 31 of RAI Responses]

The fire protection system can be aligned as a source of core injection. The fire protection pumps are diesel-driven. The alignment of the fire protection system for core injection requires several hours to accomplish. The analysis took credit for this cooling method, though apparently not for station blackout sequences. The licensee notes that this cooling method would be of minimal value in station blackout sequences because the ADS SRVs will likely reclose after battery depletion, with a consequential rise in reactor pressure that would make fire protection injection unavailable. [pp. 10, 11 of RAI Responses, pp. 3-32, 3-89, 6-11, 6-12 of submittal]

The control rod drive (CRD) system was modeled as a source of flow to the reactor vessel. The CRD flow rate at a 1,000 psig reactor pressure is about 140 gpm with one pump, and 150 gpm with two pumps. A flow rate of 140 gpm was used in the analysis. MAAP simulations performed by the licensee indicate that CRD with one pump running (140 gpm at 1,000 psig) is adequate after one hour to avert core damage. [p. 11 of RAI Responses, pp. 3-30, 3-32, 3-80, 3-182 of submittal]

In the ISLOCA analysis, no credit was taken for mitigating systems in which the ISLOCA occurred. Each of the ECCS systems is located in its own flood-proof room which prevents flood waters from traveling from the area where the break occurred to other ECCS rooms. Because the ECCS pump rooms are not vapor tight, steam can be transported among these rooms. However, the ECCS equipment qualification envelop demonstrates the operability of ECCS equipment after exposure to high temperature and humidity conditions. The IPE assumes that an ISLOCA will not depressurize the reactor to the point where low pressure injection systems can provide makeup. However, credit was taken for use of the low pressure injection systems in conjunction with operator depressurization. [p. 8 of RAI Responses, p. 3-57 of submittal]

The ISLOCA analysis does not include the possibility of break isolation. It is assumed that ECCS systems can provide adequate makeup for the 24 hour accident mission time. The possibility of suppression pool depletion during the mitigation period was not addressed in the IPE. A 1993 (post-IPE) study examined the likelihood of suppression pool depletion through the two predominant ISLOCA paths, the RHR shutdown cooling line and the feedwater lines. This study estimated that the frequency of an inventory-depleting ISLOCA event through the RHR shutdown cooling line has a frequency of 3.3E-08/yr. For the feedwater lines, the frequency of an inventory-depleting ISLOCA was estimated to be 2.3E-08/yr. In other words, the total frequency of an inventory-depleting ISLOCA was estimated to be 5.6E-08/yr. All of the IPE ISLOCA sequences were below the truncation valve of 1.1E-09/yr. Therefore, had the IPE considered suppression pool inventory depletion, the ISLOCA frequency would not exceed 5.6E-08/yr. [p. 8 of RAI Responses, pp. 3-57, 4-33, 4-79 of submittal]

Like the Grand Gulf NUREG/CR-4550 study, the Clinton IPE took credit for HPCS as an ATWS mitigating system. [pp. 3-27, 3-59 to 3-62 of submittal]

2.2.3 Systems Analysis.

Systems descriptions are included in Section 3.2.1 of the submittal. These system descriptions contain information on system function, system design and operation, modeling assumptions, operator actions, and system interfaces. The system descriptions also contain simplified schematics that show major equipment items and important flow and configuration information. [pp. 3-63 to 3-161 of submittal]

Clinton has two turbine-driven reactor feedwater pumps and one motor-driven feedwater pump. The motor-driven feedwater pump can supply water to the reactor regardless of the availability of motive steam and the main condenser, which are required for operation of the turbine-driven feedwater pumps. Thus, the feedwater system can provide core cooling for transients with and without main steam line isolation. [pp. 3-98, 6-3, 6-4 of submittal]

Clinton has a steam-driven RCIC system and motor-driven HPCS system, both of which are typical of BWR 6 plants. The HPCS injects over the core as opposed to the downcomer. Clinton also has a typical RHR and LPCS arrangement. The RHR system provides LPCI, as well as containment spray and suppression pool cooling. Two trains of the RHR system, "A" and "B", can operate in four different modes, specifically: LPCI, containment spray, suppression pool cooling, and shutdown. The third train of the RHR system can only operate in the LPCI mode. The LPCI injects into the core region. Spray over the top of the core can be provided by the LPCS. [p. 3-75 of submittal]

Clinton is equipped with a total of 16 safety relief valves (SRVs), 7 of which are automatically actuated by the ADS. Compressed air for the operation of these valves is required to be between 140 and 200 psig to ensure successful operation. This air is normally provided by the instrument air system. Air amplifiers are provided to boost the pressure in the instrument air system from 120 psig to a minimum of 150 psig. A backup supply of air is provided via compressed air bottles for the nine SRVs that do not have an ADS function[7]. The motor operated isolation valves to these bottles can be opened from the control room. [pp. 6.3-49, 9.3-3 of UFSAR, 3-18, 3-77 to 3-79 of submittal]

Three diesel generators are provided, one each for Class 1E electrical Divisions 1, 2, and 3, respectively. The Division 3 diesel is smaller than the other two diesels, as it provides power only to the HPCS and its required support loads. No cross-connections between Division 3 and Divisions 1 or 2 are displayed in Figure 3.2-34 of the submittal. It appears that the IPE did not take credit for using the Division 3 diesel generator to power any Division 1 or 2 equipment. The NUREG/CR 4550 PRA for Grand Gulf credited use of Division 3 power (HPCS diesel generator) to power electrical loads in Divisions 1 or 2 by means of a cross-tie. [pp. 8.3-4 of UFSAR, 3-83, 3-139 of submittal]

The shutdown service water (SX) system provides cooling water to safety related equipment when the normal balance of plant (BOP) systems are not available. During normal plant operation, the SX system is in standby while the plant service water (WS) system provides flow to various safety and non-safety related loads. Upon receipt of a LOCA signal, the SX pumps will start and the WS/SX cross tie valves close. The SX pumps will also start on receipt of a low header pressure signal, for example after a LOSP condition that would cause the WS pumps to become unavailable. [pp. 3-85, 3-86 of submittal]

The shutdown service water system can provide up to 1,000 gpm to the reactor via the RHR system when the reactor is depressurized below 50 psig. Achieving this flow rate would require isolation of all other heat loads except diesel generator cooling and the control room heating, ventilating, and air conditioning (HVAC) heat exchangers. A requirement for heat load isolation is not presently incorporated in the Clinton

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[7] Each of the SRVs, including those without an ADS function, also has an air accumulator. However, no credit was taken for these air accumulators, as their capacity was assumed insufficient for the required mission time.

procedures. Consequently, this method of core cooling was not modeled in the IPE. [p. 3-31 of submittal]

2.2.4 System Dependencies.

The submittal contains dependency matrices that identify asymmetries and include dependencies related to electrical power, instrument air, HVAC, and pump cooling. These dependency matrices contain footnotes that provide additional supporting information. [p. 3-95 of submittal]

Control room HVAC was not modeled as a required support system during the post-accident mitigating system period. The licensee states that control room HVAC is a continuously running, redundant system with a probability of failure significantly less than failure probabilities of front-line systems. Also, the large volume of the control room would lead to a relatively slow heat-up, thereby allowing additional response time for using remote shutdown capabilities. An analysis performed in response to the station blackout rule determined that the control room would not exceed 120 deg F within four hours. Procedures and equipment are also in place to provide alternate cooling measures. [p. 4 of RAI Responses]

The IPE assumed that ECCS and RCIC equipment would remain operable for 4 hours without HVAC. This assumption was based on a heatup analysis of the LPCS room and review of ECCS equipment qualification limits. HVAC unavailabilities beyond 4 hours were assumed to fail the associated ECCS/RCIC pumps. In circumstances where ECCS/RCIC pump failure occurred due to loss of HVAC (4 hours), credit was taken for backup core cooling from the CRD or fire protection system. There is no automatic trip of ECCS pumps on high temperature. [pp. 10, 11 of RAI Responses]

The RHR and LPCS pumps will continue to run for a period of time after shutdown cooling water supplies to the pump motor lube oil coolers is lost. However, the IPE did assume that the RHR and LPCS pumps will fail if lube oil cooling is lost. [p. 10 of RAI Responses, pp. 3-93, 3-104, 3-105 of submittal]

In summary, it appears that the IPE has accounted for all system dependencies.

2.3 Quantitative Process

This section of the report summarizes our review of the process by which the IPE quantified core damage accident sequences. It also summarizes our review of the data base, including consideration given to plant-specific data, in the IPE. The uncertainty and/or sensitivity analyses that were performed were also reviewed.

2.3.1 Quantification of Accident Sequence Frequencies

The IPE used a small event tree/large-fault tree technique with fault tree linking to quantify core damage sequences. The Cut Set and Fault Tree Analysis (CAFTA) code was used for the development and linking of system fault trees, and manipulations of cutsets developed from the fault trees. The Set Equation Transformation System (SETS) code was used to generate the sequence cut sets and numerical frequencies. The cut set truncation limit used for accident sequence cut sets was 1.1E-09/yr. [pp. 1-5 to 1-7, 3-46 to 3-62, 3-91 to 3-95, 3-187, 3-188 of submittal]

Credit was taken for recovery of offsite power in the IPE. Non-recovery data for LOSP were generated from information contained in NUREG-1032. The IPE non-recovery data are more optimistic that average industry experience reported in an Electric Power Research Institute (EPRI)-sponsored study [NSAC 147]. For example, at two hours, the IPE probability for non-recovery of LOSP is about a factor of 4 lower than the corresponding NSAC data. At four and eight hours, the IPE non-recovery data are approximately a factor of 5 lower than the NSAC data. [p. 12 of RAI Responses, pp. 3-180, 3-181, 3-204 of submittal]

While diesel generator failures can occur randomly during the 24 hour front-end mission time, the probability of non-recovery of offsite power significantly decreases as a function of time. Therefore, if diesel generator "run" failure rates are simply multiplied by the 24 hour front-end mission time with no further numerical adjustment, the resulting analysis is expected to be pessimistic. To more accurately account for this aspect of the analysis, the licensee applied a time-phased to station blackout cut sets. The approach used in the Clinton IPE effectively reduces the mission time used to quantify diesel generator "run" failures from 24 hours to 5 hours or less. A similar time-phased recovery analysis was also applied to diesel fuel oil pumps, and included consideration of both LOSP non-recovery probabilities and the 2 hour diesel day tank capacity. Also, a special "containment" time-phased recovery was used in the back-end analysis to account for available times to prevent vessel failure. The time-phased power recovery technique used in the Clinton IPE appears to be consistent with similar approaches used in some other IPE/PRA studies. [pp. 11 to 22 of RAI Responses, pp. 3-53, 3-180, 3-181, 3-204, 3-205, 3-201 of submittal]

The IPE credited local repair of various equipment items and systems, including diesel generators, pumps, valves, and instrumentation. It is positive that the licensee has attempted to credit a variety of repair activities to reflect the operation of the as-built, as-operated plant. However, IPE/PRA studies typically limit credit for local equipment repair activities to diesel generators, as there is comparatively more experience for repair of diesel generators than for other components and systems. It is further noted that the Clinton IPE has taken credit for up to two component/system repair actions per accident sequence cut set. Credit for multiple repair activities within a given cut set is also not typically done in IPE/PRA studies. The licensee states that the credited repair activities included in the Clinton IPE are based on a demonstrated capability of the plant to field multiple repair teams during actual emergency exercises. However, the quantification of repair activities is based on a generic EPRI database [EPRI 3000 34], and it is not clear how accurately the generic EPRI data would reflect the Clinton plant during an actual accident condition given the uncertainties inherent in predicting human actions. It is also noted that the IPE data for non-recovery of common cause diesel generator failures are one to two orders of magnitude lower (more optimistic) than industry experience used in the Accident Sequence Evaluation Program (ASEP) as reported in NUREG/CR-4550. [pp. 44 to 51 of RAI Responses, Table 8.2-10 of NUREG/CR Methodology, Vol. 1]

The selection of available time for component/system repair was based on the role of each particular system or component in preventing core damage. For injection system components failed due to loss of room cooling, a repair time of 4 hours was used. This repair time was based on a heatup analysis of the LPCS room and review of ECCS equipment qualification limits. Where injection components failed for reasons other than loss of room cooling, injection component repair appears to have been allowed only for transients. The repair time in this case was ½ hour, based on MAAP calculations that show vessel makeup can be delayed for at least ½ hour without significant core damage. For diesel generators, recovery probabilities were determined for 1 and 4 hour time periods. The 1 and 4 hour periods correspond to the times considered in the event tree for AC power recovery in time to prevent battery depletion. Recovery times for fans and shutdown service water system components were assumed to be four hours. It appears that in all cases, the maximum time analyzed in the EPRI database was only two hours. Thus, where component/system recovery could be credited for 4 hours in the IPE, 2 hour EPRI data were used. Most of the non-recovery probabilities for credited component/system repair actions range from 0.3 to 0.8. [pp. 44 to 51 of RAI Responses, pp. 3-179, 3-180 of submittal]

As part of a response to an NRC Staff request for clarification of equipment repair models, the licensee performed a sensitivity analysis that involved removal of all credited equipment repair recoveries, except those involving off-site power, DC power, and operator actions done from the control room. With this model change, the baseline CDF (exclusive of internal flooding[8]) increased by a factor of 1.44 (from 2.49E-05/yr to 3.57E-05/yr). The relative contributions of individual accident sequences were not significantly altered as a result of this sensitivity study. In no instances were increases in individual accident sequence frequencies greater than 2.4. While two new sequences were introduced as a result of the sensitivity analysis, their frequencies were less than 1E-08/yr.

While the Clinton IPE equipment repair model is more optimistic that repair models typically used in other IPE/PRA studies, the licensee's sensitivity analysis demonstrates that this repair model has not significantly affected the CDF or accident

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[8] No credit was given for flood-related operator mitigating actions

sequence profile. Therefore, it is our judgment that the licensee's equipment repair model does not represent a weakness of the IPE.

Finally, credit was taken for rapid recovery of a loss of feedwater initiating event. Quantification of this recovery action was also based on EPRI data [EPRI 3000 34]. It appears that the licensee has used a value of 0.21 as the non-recovery probability for this activity. [p. 3-180 of submittal]

2.3.2 Point Estimates and Uncertainty/Sensitivity Analyses.

The submittal does not state the statistical significance of the initiating event and fault tree basic events. However, the IPE used pipe break frequencies from WASH-1400 that represent mean values. In addition, generic transient initiator frequencies are based on mean value data provided in the NUREG/CR-4550 Grand Gulf study. Also, most of the component failure data used in the IPE analysis were extracted from the mean values presented in the NUREG/CR-4550 methodology document. The overall CDF is presented in terms of a point value. No statistical uncertainty analysis of the results was performed. [pp. 3-11, 3-13, 3-192 to 3-194 of submittal]

The licensee performed several types of sensitivity analysis. In one analysis, all recovery actions assigned a value less than 0.1 were set to 0.1. The frequency of a loss of feedwater sequence increased by a factor of 5.8, while the frequencies of several other sequences increased by factors less than two. The overall CDF increased by only 4%. [pp. 3-182, 3-183 of submittal]

The licensee performed sensitivity analyses related to the CDF impact from two plant improvements. These sensitivity analyses are summarized in Subsection 2.7.3 of this report. As previously discussed in Subsection 2.3.1 of this report, the licensee also generated a sensitivity analysis related to the IPE component/system repair models.

A sensitivity analysis was performed on preliminary front-end results to identify human events requiring possible refinement of their quantification. Following this screening process, some of the human event data were re-quantified before the final results were generated. A set of back-end MAAP sensitivity analyses was also performed. [pp. 3-172, 3-173, 3-198 to 3-200, 4-50 to 4-65 of submittal]

2.3.3 Use of Plant-Specific Data.

Plant-specific data were used to quantify maintenance unavailabilities. However, all of the component hardware failure rates, with the possible exception of diesel generator start failures, were quantified with generic data[9]. The licensee states that the short time history of the plant (approximately six years at the time the IPE was performed) is unlikely to provide sufficient failure data to support the analysis. The licensee further states that plant-specific data were not ignored, even though the decision was made to use generic data for component hardware failures. The licensee examined plant-specific data to determine if any unusual failure rates were occurring as summarized below: [pp. 1, 2 of RAI Responses, pp. 3-164 to 3-167, 3-193, 3-207 of submittal]

- Even at the time of the IPE, various safety system out-of-service times compared favorably with the other domestic BWR-6 plants.

- To date, there have been no surveillance-related pump start failures on any of the safety-related injection systems. Based on quarterly testing, this result corresponds to no failures in at least 157 attempts, or a start failure probability less than 6E-03. The generic value used in the IPE for the probability of pump start failure was 3E-03.

- Other than the HPCS water leg pump, there have been only 2 pump "run" failures in about 200,000 hours. One of the failures was a post-maintenance test failure and was not counted. By counting the remaining pump failure, a pump failure-to-run rate is 5E-06/hr, compared to a generic value of 5E-05/hr. The HPCS water leg pump failures were attributed to a design problem and no subsequent failures have occurred following corrective actions.

- Of a population of 170 safety-related and risk significant motor-operated valves (MOVs), there have been 37 cases recorded as valve failures. Assuming that each valve is stroked only once per quarter (minimum surveillance requirements), this operating experience supports a failure-to-stroke rate of 7E-03 per demand. The failure probability of 7E-03 is pessimistic, because some of the recorded failures include non-risk significant failures such as seat leakage. The IPE used a generic value of 3E-03 for MOV demand failure probabilities.

- The average forced outage rate over the commercial life of the plant has been about 3.5 outages per operating year, including the first years of plant life, as compared to a generic frequency of 7 events per year.

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[9] The licensee states that the diesel generators have been started a sufficient number of times (306) during surveillance testing to determine a plant-specific failure rate estimate. The plant-specific diesel generator start failure probability, 2.0E-02, is close to the NUREG/CR-4550 generic estimate of 3E-02. It is unclear, however, if the IPE actually used the plant-specific estimate. For example, common cause data listed in Table 3.3-12 of the submittal suggests that the licensee used the generic estimate of 3E-02. [pp. 3-165, 3-166, 3-193, 3-207 of submittal]

Based on the above discussion, the licensee has demonstrated instances where plant-specific failure data are comparable to or better than corresponding generic data. However, in our judgment, the omission (or very limited use) of plant-specific hardware data represents a weakness of the analysis. While it might be argued that the use of generic data provides an upper bound to the total CDF, the relative CDF contributions of various sequences and failure events may be distorted.[10]

As previously noted in Section 2.2.1 of this report, plant data were generally omitted from the development of initiating event frequencies.

2.3.4 Use of Generic Data.

The primary source of generic data was the NUREG/CR-4550 methodology document. The other sources of generic data were the NUREG/CR 4550 Grand Gulf study, NUREG/CR 2815, IEEE 500, and (unspecified) General Electric reliability data reports. [p. 22 of RAI Responses, pp. 3-146, 3-165, 3-192 to 3-194 of submittal]

We performed a comparison of IPE generic component failure data to generic values used in NUREG/CR-4550. This comparison is presented below in Table 2-1.

Table 2-1. Generic Component Failure Data[1]

Component	Failure Mode	IPE Mean Value Estimate	NUREG/CR-4550 Mean Value Estimate
Turbine-driven pump	Start	3E-02	3E-02
	Run	5E-03 (first hour) 2E-05 (subsequent hours)	5E-03
Motor-driven pump (see note 2)	Start	3E-03	3E-03
	Run	3E-05	3E-05
MOV	Transfer	3E-03	3E-03
Check valve	Open (demand)	1E-04	1E-04
Battery charger	No output	1E-06	1E-06
Battery	No output	1E-06	1E-06
Inverter	No output	1E-04	1E-04
Circuit breaker	Transfer	3E-03	3E-03
Diesel generator	Start	3E-02	3E-02
	Run	2E-03	2E-03
Strainer/Filter	Plugs	3E-05	3E-05
Transformer	No output (short/open)	2E-06	2E-06

Notes: (1) Failures to start, open, close, operate, or transfer are probabilities of failure on demand. The other failures represent frequencies expressed per hour. (2) IPE data used for various motor-driven pumps, including diesel fuel oil pumps.

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[10] The licensee states that plant-specific component failure data will be included in future updates to the IPE as "statistically valid" data are gathered.

Table 2-1 shows that IPE generic data are consistent with NUREG/CR-4550 data, except for the turbine-driven pump run data. The IPE has used a value of 2E-05/hr for a turbine-pump to run after the first hour, compared with the NUREG/CR-4500 value of 5E-03/yr applicable for all run periods. This element of the IPE failure data was extracted from the PRA Procedures Guide [NUREG/CR 2815].

As was previously noted in Section 2.2.1 of this report, generic industry data were used in the development of LOCA and transient initiating events.

2.3.5 Common-Cause Quantification.

The estimation of common-cause failure probabilities was based on the Multiple Greek Letter (MGL) method. A number of component groups were included the common cause analysis, including diesel generators, pumps, MOVs, AOVs, check valves, explosive valves, circuit breakers, battery chargers, inverters, relays, safety relief valves, batteries and instrumentation and control components. The submittal does not state the source of the MGL data used in the IPE. [pp. 2-4, 2-5, 3-185, 3-206 to 3-208 of submittal]

We performed a comparison of IPE common-cause data with generic beta factors used in the NUREG/CR-4550 studies [NUREG/CR 4550, Methodology]. In preparing this comparison, the MGL-based failure rates provided in Table 3.3-12 of the submittal were used to derive equivalent fractional failures to correspond to the beta factors presented in NUREG/CR-4550. The common cause data comparison is summarized below in Table 2-2. [pp. 3-207, 3-208 of submittal]

Table 2-2. Comparison of IPE and NUREG/CR-4550 Common-Cause Data

Component	Failure Mode	Group Size	Equivalent IPE Beta Factor Derived from Table 3.3-12 of Submittal	NUREG/CR 4550 Mean Value Beta Factor
Shutdown Service Water Pump	Start	2	0.17	0.026
	Run	2	0.067	
RHR/LPCS Pump	Start	4	0.037	0.10
	Run	4	0.04	
MOV	Transfer	2	0.33	0.088
		4	0.0037	0.057
AOV	Transfer	2	0.15	0.10 (2 or more)
Diesel Generator	Start	3	0.0097	0.018
	Run	3	0.042	

As indicated in Table 2-2, the IPE common cause data for start failure of 2 shutdown service water pumps is over a factor of 6 higher than the NUREG/CR-4550 generic data. Also, the IPE common cause data for failure of 2 MOVs is almost a factor of 4 higher than the generic data. On the other hand, the IPE data for common cause failure of 4 MOVs is about a factor of 15 lower than the generic data. In addition, the IPE data for RHR/LPCS pump and diesel generator start failures are lower than generic data by factors of about 3 and 2, respectively.

The licensee states that a beta factor of 0.02 was used to quantify common cause failures of diesel fuel oil pumps. This value is based on NUREG/CR-2098. [p. 22 of RAI Responses]

2.4 Interface Issues

This section of the report summarizes our review of the interfaces between the front-end and back-end analyses, and the interfaces between the front-end and human factors analyses. The focus of the review was on significant interfaces that affect the ability to prevent core damage.

2.4.1 Front-End and Back-End Interfaces.

The IPE assumes that the LPCS, the HPCS, and the RHR pumps (in the LPCI mode) do not lose suction after loss of containment heat removal or containment depressurization following containment venting or containment failure unless the failure is in the suppression pool. Per design, sufficient NPSH is expected to remain available to operate these pumps with the suppression pool at saturation conditions. [p. 60 of RAI Responses, pp. 6.3-5, 6.3-12, 6.3-17 of UFSAR, 3-28 of the submittal]

As long as the reactor is shutdown and core damage is averted via ECCS cooling, loss of containment cooling will not cause containment failure during the 24 hour front-end mission time. The licensee states that containment cooling is not required due to the relatively large suppression pool volume and free air volume. Because ECCS systems would not be affected during the front-end mission time, containment venting was not required or modeled in the front-end analysis. [p. 31 of RAI Responses]

While six vent paths containment vent paths are available, only the three with the largest capacity were modeled in the back-end analysis. The other three paths do not have sufficient capacity by themselves to vent containment. The modeled paths are: the RHR system through the fuel pool cooling and cleanup (FC) system; the FC system through the spent fuel pool; and through a hole cut in the piping of the containment continuous purge systems. One of the credited paths vents directly to the outside so that operator access to plant areas is minimally impacted. The other two paths are scrubbed through the spent fuel pool to minimize the impact on area accessibility. [p. 60 of RAI Responses, pp. 3-22, 3-80, 3-81 of submittal]

The possibility of suppression pool depletion during an ISLOCA was not addressed in the IPE. A 1993 (post-IPE submittal) study examined the likelihood of suppression pool depletion through the two predominant ISLOCA paths, the RHR shutdown cooling line and the feedwater lines. This study estimated that the frequency of an inventory-depleting ISLOCA event through the RHR shutdown cooling line has a frequency of 3.3E-08/yr. For the feedwater lines, the frequency of an inventory-depleting ISLOCA was estimated to be 2.3E-08/yr. In other words, the total frequency of an inventory-depleting ISLOCA was 5.6E-08/yr. All of the IPE ISLOCA sequences were below the truncation valve of 1.1E-09/yr. Therefore, had the IPE considered suppression pool inventory depletion, the ISLOCA frequency would not exceed 5.6E-08/yr. [p. 8 of RAI Responses, pp. 3-57, 4-33, 4-79 of submittal]

The Level 1 core damage end states were binned into accident classes to form the beginning states for containment event trees. The binning of Level 1 end states into accident classes was based on the following criteria: containment integrity, primary system integrity, relative timing of core damage, primary system pressure, and failure of critical functions leading to core damage. These five classes were further subdivided into subclasses based on the unavailability of key functions. The binning process used to couple core damage sequences into the back-end analysis appears to be comparable with the process used in other PRA/IPE studies. [pp. 2-6, 2-7, 3-34, 3-35, 4-22, 4-23 of submittal]

2.4.2 <u>Human Factors Interfaces</u>.

Dominant human errors and recovery factors contributing to CDF include: [p. 6-27 of submittal]

- Failure to recover offsite power in 0.5 hours
- Operator fails to manually initiate ADS

Operator failure to manually initiate ADS for use of low pressure injection contributes about 24% to the total CDF. In contrast, equipment failures related to ADS failure are relatively minor contributors to CDF. The licensee states that per procedure, there are no conditions were automatic depressurization would be allowed. At the same time, the Clinton procedures are stated to be consistent with Revision 4 of the BWR Emergency Procedure Guidelines (EPGs). The failure of operators to manually initiate ADS was quantified with a human error probability (HEP) of 5E-04. Alternate methods of vessel depressurization were not credited in the IPE (for example, via MSIVs and turbine bypass valves). As stated by the licensee, failure of operators to depressurize does not represent a vulnerability because this action has been emphasized in training and has been judged unlikely to induce operator error. [pp. 26 to 30 of RAI Responses]

As previously discussed in Subsection 2.3.1 of this report, the IPE took credit for component/system repair recoveries, including up to two such recoveries per accident sequence cut set. Credit was also taken for rapid recovery of a loss of feedwater initiating event. [p. 3-180 of submittal]

2.5 Evaluation of Decay Heat Removal and Other Safety Issues

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

2.5.1 Examination of DHR.

The DHR contribution to CDF was derived by eliminating from accident sequence cutsets failures of systems that cannot remove decay heat. Systems not able to remove decay heat include HPCS, RCIC, LPCS, ADS, and fire protection. The CDF due to loss of DHR was estimated to be 5.2E-06/yr. The licensee states that this DHR-related CDF estimate is pessimistic, as additional methods of DHR were not credited, for example RHR lined up to the fuel pool cooling and cleanup system. [p. 3-231 of submittal]

As pointed out by the licensee, the USI A-45 study recommends that DHR-related CDF contributions should not be greater than 1E-05/yr. The Clinton DHR-related CDF was determined to be no greater than 5.2E-06/yr. No DHR-related vulnerabilities were identified. [p. 3-232 of submittal]

2.5.2 Diverse Means of DHR.

The IPE evaluated the diverse means for accomplishing DHR, including: use of the power conversion system, RCIC, HPCS, and use of low pressure injection by opening safety relief valves. [pp. 3-230, 3-231 of submittal]

2.5.3 Unique Features of DHR.

The unique features at Clinton that directly impact the ability to provide DHR are as follows: [pp. 4-1, 4-42, 4-43, 6-2 to 6-4, 6-11 to 6-14 of submittal]

- Diversity of reactor feedwater pump motive power. Two of the reactor feedwater pumps are turbine-driven, while the third pump is motor-driven. The motor-driven pump provides the capability of providing feedwater for transients with and without main steam isolation. This design feature tends to decrease the CDF.

- Ability of emergency core cooling system (ECCS) pumps to operate with a saturated suppression pool. The high pressure core spray (HPCS), low pressure core spray (LPCS), and residual heat removal (RHR) pumps can operate with a saturated suppression pool and thus provide core cooling in the event containment cooling is lost. This design feature tends to decrease the CDF.

- Ability to cross-connect the fire protection system for core injection. The fire protection system can be aligned as a source of core injection. The fire protection pumps are diesel-driven. This design feature tends to decrease the CDF. The analysis took credit for this cooling method, though apparently not for station blackout sequences. The licensee notes that this cooling method would be of minimal value in station blackout sequences because the ADS SRVs will likely reclose after battery depletion, with a consequential rise in reactor pressure that would make fire protection injection unavailable.

2.5.4 Other GSI/USIs Addressed in the Submittal.

The submittal does not address GSIs/USIs other than DHR. The licensee states that there are no open generic issues at Clinton. [p. 3-232 of submittal]

2.6 Internal Flooding

This section of the report summarizes our reviews of the process used to model internal flooding and of the results of the analysis of internal flooding.

2.6.1 Internal Flooding Methodology.

The flooding analysis considered effects from both submergence and spray. Based on a post-IPE submittal review, the licensee concluded that the analysis also accounts for steam impingement. [p. 26 of RAI Responses, p. 3-189 of submittal]

The flooding analysis evaluated events that could lead to a scram or shutdown requiring core cooling systems. Plant walkdowns, a Sargent & Lundy flooding report [S&L Flood], and input from the IPE Senior Reactor Operator were used to support the analysis. Propagation of a flood beyond the flood-initiation area through doorways, hatches, stairwells, etc. was addressed. [pp. 3-189, 3-190 of submittal]

The frequency of flooding initiating events was based on a review of the specific components that could rupture or leak and cause a flood. Random equipment failures were considered as initiating events, as well as personnel failures to perform system isolation prior to maintenance. Equipment considered in the development of initiating events included piping, expansion joints, valves and tanks. The initiating event frequencies were extracted from WASH-1400, PRAs for Seabrook and Oconee, and NUREG/CR-1363. Sequence quantification was based on the internal events logic models, with consideration given to flood-related component failures. No credit was taken for flood-related operator mitigating actions, such as flood isolation or tripping of pumps. [pp. 56, 57 of RAI Responses, pp. 3-188 to 3-191, 3-211 of submittal]

2.6.2 Internal Flooding Results.

The total point estimate CDF contribution from internal flooding was calculated to be 1.6E-06/yr, which represents about 6% of the total CDF. Five dominant sequences collectively represent 75% of the total flooding-related CDF contribution. These sequences are summarized below in Table 2-3. [pp. 3-224 to 3-228, 3-235 of submittal]

Table 2-3. Dominant Sequences From Internal Flooding Analysis

Initiating Event	CDF (per yr)
Feedwater line break in main steam tunnel	4.17E-07
Plant service water line break in CCW pump and tank area (control building elev 762')	2.24E-07
Plant service water break in HPCS pump room	2.23E-07
HPCS line rupture in HPCS pump room	1.79E-07
CCW line break in CCW pump and tank area (control building elev 762')	1.55E-07

2.7 Core Damage Sequence Results

This section of the report reviews the dominant core damage sequences reported in the submittal. The reporting of core damage sequences- whether systemic or functional- is reviewed for consistency with the screening criteria of NUREG-1335. The definition of vulnerability provided in the submittal is reviewed. Vulnerabilities, enhancements, and plant hardware and procedural modifications, as reported in the submittal, are reviewed.

2.7.1 Dominant Core Damage Sequences.

The IPE utilized event trees that are generally functional in nature, and reported results using the screening criteria from Appendix 2 of Generic Letter 88-20 for functional sequences. The total point estimate CDF for Clinton is 2.6E-05/yr, including a 1.6E-06/yr contribution from internal flooding[11]. [pp. 3-212, 3-213 of submittal]

Accident types and their percent contribution are listed in Table 2-4. [pp. 1-10, 1-11 of submittal]

[11] The most recent update of the IPE predicts a CDF exclusive of flooding of 5.5E-06/yr. [p. 5 of RAI Responses]

Table 2-4. Accident Types and Their Contribution to Core Damage Frequency

Accident Type	CDF Contribution pr yr.	Percent Contribution to CDF
Transients	1.4E-05	52
Station Blackout	9.8E-06	37
Internal Flooding	1.6E-06	6
LOCA (includes IORV)	1.1E-06	4
ATWS	1.4E-07	0.5
ISLOCA	negligible	negligible

As previously noted, the licensee defines station blackout to be loss of offsite power combined with loss of power from the Division 1 and 2 diesel generators. The status of the Division 3 diesel generator (HPCS) is not considered in the definition of station blackout. [pp. 3-4, 3-53, 6-5 of submittal]

Initiating event contributions to the CDF, and their percent contribution, are listed below in Table 2-5[12]. [pp. 1-9, 3-235 of submittal]

Seven functional sequences were identified above the Generic Letter 88-20 screening criteria of 1.0E-06/yr. These dominant functional sequences are listed in Table 2-6 of this report. [pp. 1-8, 1-14, 3-224, 3-225, 3-235 of submittal]

Operator failure to manually initiate ADS for use of low pressure injection contributes about 24% to the total CDF. In contrast, equipment failures related to ADS failure are relatively minor contributors to CDF. The licensee states that per procedure, there are no conditions were automatic depressurization would be allowed. At the same time, the Clinton procedures are stated to be consistent with Revision 4 of the BWR Emergency Procedure Guidelines (EPGs). The failure of operators to manually initiate ADS was quantified with a human error probability (HEP) of 5E-04. Alternate methods of vessel depressurization were not credited in the IPE (for example, via MSIVs and turbine bypass valves). As stated by the licensee, failure of operators to depressurize does not represent a vulnerability because this action has been emphasized in training and has been judged unlikely to induce operator error. [pp. 26 to 30 of RAI Responses]

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[12] With the exception of lower order flooding events, this table is complete. This table was assembled from information contained in Tables 1.4-1 and 3.4-3 of the submittal. Submittal Table 1.4-1 presents initiating events for non-flood events, while submittal Table 3.4-3 presents dominant flooding sequences. Together, the internal flood initiating events listed in submittal Table 3.4-3 represent about 75% of the total flood-related CDF. Because internal flooding contributes about 6% to the total CDF, the missing flood-related initiating events represent about 1.5% of the total CDF.

Table 2-5. Initiating Events and Their Contribution to Core Damage Frequency

Initiating Event	CDF Contribution/yr.	% Cont. to CDF
LOSP	1.2E-05	46
Transient without isolation	4.8E-06	18
Transient with isolation	4.2E-06	16
Loss of DC Bus	1.2E-06	4.6
Inadvertent open relief valve (IORV)	1.1E-06	4.2
Loss of feedwater	9.6E-07	3.7
Feedwater line break in main steam tunnel (internal flood)	4.2E-07	1.6
Plant service water line break in CCW pump/tank area (internal flood)	2.2E-07	0.9
Plant service water break in HPCS pump room (internal flood)	2.2E-07	0.9
Loss of service water	1.9E-07	0.7
HPCS line rupture in HPCS pump room (internal flood)	1.8E-07	0.7
CCW line break in CCW pump/tank area (internal flood)	1.6E-07	0.6
ATWS (see note 1)	1.4E-07	0.5
Medium LOCA	1.3E-08	0.05
Loss of instrument air	1.0E-08	0.04
Large LOCA	negligible	negligible
Small LOCA	negligible	negligible
ISLOCA	negligible	negligible

Notes: (1) The submittal lists ATWS as an "initiating event"; other IPE/PRA studies generally categorize ATWS as an accident type.

Table 2-6. Dominant Functional Core Damage Sequences

Initiating Event	Dominant Subsequent Failures in Sequence	% Contribution to Total CDF
LOSP	Division I and II diesel generators fail, HPCS and RCIC fail (short-term station blackout scenario)	20
LOSP	Division I and II diesel generators fail, HPCS fails, RCIC runs until battery fails (long-term station blackout scenario)	18
Transient Without Isolation	All high pressure injection fails, depressurization fails	13
Transient With Isolation	All high pressure injection fails, depressurization fails	12
Internal Flooding (combination of 5 separate initiating events)	Most significant scenario involves a feedwater line break in the steam tunnel that floods RCIC, LPCS, and RHR "A" train equipment	6
Loss of Non-Safety DC Bus	Main condenser and all injection sources fail	4
Open relief valve	Loss of feedwater delivery and all high and low pressure injection systems; in many cases, failure of injection is related to lack of ac power	4

Results from a Fussel-Vesely importance analysis were presented in the submittal. The most important events based on this measure are listed below: [pp. 6-4, 6-5, 6-27 to 6-32 of submittal]

- Failure to recover off-site power in 0.5 hours
- Loss of off-site power (initiating event)
- Independent sub-tree containing HPCS failure basic events
- Basic event representing recovery of HPCS failures
- Independent sub-tree containing RCIC failure basic events
- Basic event representing recovery of RCIC failures
- Operator fails to manually initiate ADS

Finally, as previously discussed in Subsection 2.3.1 of this report, the licensee performed a sensitivity analysis that involved removal of all credited equipment repair recoveries, except those involving off-site power, DC power, and operator actions done from the control room. With this model change, the baseline CDF (exclusive of internal flooding[13]) increased by a factor of 1.44 (from 2.49E-05/yr to 3.57E-05/yr). The relative contributions of individual accident sequences were not significantly altered as a result of this sensitivity study. In no instances were increases in individual accident sequence frequencies greater than 2.4. While two new sequences were introduced as a result of the sensitivity analysis, their frequencies were less than 1E-08/yr.

While the Clinton IPE equipment repair model is more optimistic that repair models typically used in other IPE/PRA studies, the licensee's sensitivity analysis demonstrates that this repair model has not significantly affected the CDF or accident sequence profile. Therefore, it is our judgment that the licensee's equipment repair model does not represent a weakness of the IPE.

2.7.2 Vulnerabilities.

The licensee used the following criteria to identify vulnerabilities: [p. 3-228 of submittal]

- New or unusual means by which core damage or containment failure occur as compared to those identified in other PRAs, or
- Results that suggest the plant CDF would not be able to meet the NRC's safety goal for core damage (1E-04/yr), or
- Systems, components, or operator actions that control the core damage result (i.e., greater than 90%).

As stated by the licensee, accident classes that contribute to core damage at Clinton are similar to those identified in PRAs of comparable facilities, such as the

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[13] No credit was given for flood-related operator mitigating actions

NUREG/CR-4550 Grand Gulf study. It is also stated that the CDF internal events estimate of 2.6E-05/yr leaves ample margin for accommodating risks of external events and still meet the NRC's (former proposed) safety goal of 1E-04/yr. Based on the above criteria, the licensee determined that there are no vulnerabilities at Clinton. [pp. 3-228, 3-229 of submittal]

2.7.3 Proposed Improvements and Modifications.

Several potential improvements were identified as a result of the IPE. None of these improvements was credited in the IPE version reported in the submittal. The plant improvements are summarized in Table 2-7. [pp. 4 to 7 or RAI Responses, pp. 6-1, 6-4 to 6-26 of submittal]

The licensee provided information regarding plant changes made in response to the Station Blackout Rule, and other modifications separate from the Station Blackout Rule that reduce the station blackout CDF. These modifications are summarized in Table 2-8. [pp. 9, 10 of RAI Responses]

Table 2-7. Summary of Plant Improvements

Plant Improvement	Status	Notes	Estimated CDF Impact
Improvements Affecting Core Damage Risk			
Operator training to emphasize importance of maintaining off-site power	Complete		Not available
Operator training to emphasize importance of manual ADS initiation	Complete		Not available
Modify HPCS surveillance procedure to demonstrate unobstructed flow path from suppression pool	Complete	CDF reduction based on IPE reported in submittal (CDF reduced from 2.6E-05/yr to 2.3E-05/yr)	12.8% reduction (see note at left)
Install bypass line to allow easier use of fire protection system for vessel makeup	Deferred (see note 1 at right)	(1) Licensee has not yet decided whether to make this modification; decision will be based on cost-benefit analysis (2) For IPE reported in submittal, 13% CDF reduction (from 2.6E-05/yr to 2.3E-05/yr); for latest IPE update, 9% CDF reduction (from 5.5E-06 to 5.0E-06/yr)	13% reduction (see note 2 at left)
Evaluate possible changes to training program beneficial to recover AC power supplies during LOSP	Complete	It is not clear what (if any) changes were made as a result of this evaluation	Not available
Provide additional procedural confirmation that shutdown service water pumps have started when required for diesel generator operation	Dropped (see note at right)	Modification not made due to small perceived benefit	Not available
Improvements Affecting Back-End Risk			
Operator training to emphasize importance of maintaining off-site power related to preventing offsite releases	Complete		Not applicable
Operator training to emphasize importance of AC power recovery	Complete		Not applicable
Operator training to emphasize importance of manually isolating containment bypass path into fuel pool cooling/cleanup line during station blackout	Complete	Required isolation accomplished by closing valve 1FC008	Not applicable
Operator training to emphasize significance of scram system hardware failures as related to release frequencies	Complete		Not applicable

Table 2-8. Summary of Plant Changes Directly Related to Station Blackout

Description of Plant Change	Status	Plant Change Accounted for in IPE?	Notes	Estimated CDF Impact
Modifications Specifically Related to Station Blackout Rule				
Procedures for DC load shedding during station blackout	Complete	Yes	Apparently implemented after Dec 31, 1991 IPE freeze date	Not available
Procedures for RCIC and HPCS operation during station blackout	Complete	Yes	Apparently implemented after Dec 31, 1991 IPE freeze date	Not available
Portable fan to cool main control room during station blackout	Complete	No		Not available
Modifications Separate From Station Blackout Rule				
Installation of concrete barriers around all outside transformers	Complete	No	Barriers protect transformers from damage due to vehicle or failure of adjacent transformer	Not available

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

This section of the report provides an overall evaluation of the quality of the IPE based on this review. Strengths and weaknesses of the IPE are summarized. Important assumptions of the model are summarized. Major insights from the IPE are presented.

Because Clinton began commercial operation in April 1987, there is a relatively limited operational history from which to derive plant-specific failure rates. While plant-specific data were used in the IPE for test and maintenance unavailabilities, component hardware failures were entirely based on generic data (with the possible exception of diesel generator start failures). Initiating event frequencies were for the most part also based on generic data. To support this wide use of generic data, the licensee cited instances where plant-specific failure data are comparable to or better than corresponding generic data.

Some other plants with limited operational experience have used plant-specific data to update generic data via a Bayesian process. It is not clear why the Clinton IPE did not use a similar approach. In our judgment, the limited use of plant-specific data represents a weakness of the Clinton analysis. While it might be argued that the use of generic data provides an upper bound to the total CDF, the relative CDF contributions of various sequences and failure events may be distorted.

It is also noteworthy that the Clinton IPE credited local repair of various equipment items and systems, including diesel generators, pumps, valves, and instrumentation. It is positive that the licensee has attempted to credit a variety of repair activities to reflect the operation of the as-built, as-operated plant. However, IPE/PRA studies typically limit credit for local equipment repair activities to diesel generators, as there is comparatively more experience for repair of diesel generators than for other components and systems. It is further noted that the Clinton IPE has taken credit for up to two component/system repair actions per accident sequence cut set. Credit for multiple repair activities within a given cut set is also not typically done in IPE/PRA studies. The licensee states that the credited repair activities included in the Clinton IPE are based on a demonstrated capability of the plant to field multiple repair teams during actual emergency exercises. However, the quantification of repair activities is based on a generic EPRI database, and it is not clear how accurately the generic EPRI data would reflect the Clinton plant during an actual accident condition given the uncertainties inherent in predicting human actions. It is also noted that the IPE data for non-recovery of common cause diesel generator failures are one to two orders of magnitude lower (more optimistic) than industry experience used in the Accident Sequence Evaluation Program (ASEP) as reported in NUREG/CR-4550 (Rev. 1, Methodology).

As part of a response to an NRC Staff request for clarification of equipment repair models, the licensee performed a sensitivity analysis that involved removal of all credited equipment repair recoveries, except those involving off-site power, DC power, and operator actions done from the control room. With this model change, the baseline CDF (exclusive of internal flooding[14]) increased by a factor of 1.44 (from 2.49E-05/yr to 3.57E-05/yr). The relative contributions of individual accident sequences were not significantly altered as a result of this sensitivity study. In no instances were increases in individual accident sequence frequencies greater than 2.4. While two new sequences were introduced as a result of the sensitivity analysis, their frequencies were less than 1E-08/yr.

While the Clinton IPE equipment repair model is more optimistic that repair models typically used in other IPE/PRA studies, the licensee's sensitivity analysis demonstrates that this repair model has not significantly affected the CDF or accident sequence profile. Therefore, it is our judgment that the licensee's equipment repair model does not represent a weakness of the IPE.

Significant level-one IPE findings are as follows:

- Operator failure to manually initiate ADS for use of low pressure injection contributes about 24% to the total CDF. The licensee does not consider failure of operators to depressurize as a vulnerability because this action has been emphasized in training and has been judged unlikely to induce operator error.

Based on this review, the following aspect of the IPE modeling process has an impact on the overall CDF:

- The HPCS, LPCS, and RHR pumps can operate with a saturated suppression pool and thus provide core cooling in the event containment cooling is lost. This design feature tends to decrease the CDF.

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[14] No credit was given for flood-related operator mitigating actions.

4. DATA SUMMARY SHEETS

This section of the report provides a summary of information from our review.

Initiating Event Frequencies

Initiating Event		Frequency Per Year
Small Break LOCA		1.00E-03
Medium Break LOCA		3.00E-04
Large Break LOCA		1.0E-04
Interfacing LOCA (see breakdown below)		5.00E-06
Breakdown of ISLOCA IE		
LPCI Injection Lines	1.47E-07	
LPCS Injection Line	2.86E-08	
Shutdown Cooling Suction Line	2.54E-06	
RPV Head Spray Line	4.94E-11	
HPCS Line	1.98E-09	
Feedwater Lines	2.28E-06	
Shutdown Cooling Return Lines	3.31E-11	
Inadvertent/Stuck-Open Safety Relief Valve (IORV)		1.00E-01
Loss of Offsite Power		8.4E-02
Loss of Feedwater		.06
Transient with Isolation		1.7
Transient without Isolation		4.7
Loss of Instrument Air		4.32E-03
Loss of Service Water		1.75E-03
Loss of Non-Safety DC Bus		1.39E0-02

Overall CDF

The total point estimate CDF for Clinton is 2.6E-05/yr, including internal flooding. The CDF contribution from flooding is 1.6E-06/yr.

Dominant Initiating Events Contributing to CDF

Loss of off-site power	46%
Transient w/o isolation from main cond.	18%
Transient with isolation from main cond.	16%
Loss of DC bus	5%
Inadvertent Open Relief Valve (IORV)	4%

Loss of Feedwater	4%

Dominant Hardware Failures and Operator Errors Contributing to CDF

Dominant hardware failures contributing to CDF include:

- Failure to recover off-site power in 0.5 hours
- Independent sub-tree containing HPCS failure basic events
- Basic event representing recovery of HPCS failures
- Independent sub-tree containing RCIC failure basic events
- Basic event representing recovery of RCIC failures

Dominant human errors and recovery factors contributing to CDF include:

- Failure to recover offsite power in 0.5 hours
- Operator fails to manually initiate ADS

Dominant Accident Classes Contributing to CDF

Transients	5[illegible]%
Station blackout	37%
Internal Flooding	6%
LOCA (includes IORV)	4%
Anticipated transient without scram (ATWS)	0.5%
Interfacing Systems LOCA (ISLOCA)	negligible

Design Characteristics Important for CDF

The following design features impact the CDF:

- Four hour battery lifetime. With credit for load shedding, the battery lifetime can be extended to 4 hours. However, a 4 hour battery lifetime is less than battery lifetimes at some other BWRs. The 4 hour battery lifetime at Clinton tends to increase the CDF compared to those BWRs with longer battery lifetimes.

- Ability of emergency core cooling system (ECCS) pumps to operate with a saturated suppression pool. The high pressure core spray (HPCS), low pressure core spray (LPCS), and residual heat removal (RHR) pumps can operate with a saturated suppression pool and thus provide core cooling in the event containment cooling is lost. This design feature tends to decrease the CDF.

- Ability to cross-connect the fire protection system for core injection. The fire protection system can be aligned as a source of core injection. The fire protection pumps are diesel-driven. This design feature tends to decrease the

CDF. The analysis took credit for this cooling method, though apparently not for station blackout sequences. This cooling method would be of minimal value in station blackout sequences because the ADS SRVs will likely reclose after battery depletion, with a consequential rise in reactor pressure that would make fire protection injection unavailable.

Modifications

The following plant improvements were identified as a result of the IPE:

Front-end

- Operator training to emphasize importance of maintaining off-site power
- Operator training to emphasize importance of manual ADS initiation
- Modify HPCS surveillance procedure to test suppression pool suction path
- Install bypass line to allow easier use of fire protection system for vessel makeup

Back-end

- Operator training to emphasize importance of maintaining off-site power to prevent offsite releases
- Operator training to emphasize importance of AC power recovery
- Operator training to emphasize importance of manually isolating containment bypass path into fuel pool cooling/cleanup line during station blackout
- Operator training to emphasize significance of scram system hardware failures as related to release frequencies

Other USI/GSIs Addressed

None.

Significant PRA Findings

- Operator failure to manually initiate ADS for use of low pressure injection contributes about 24% to the total CDF. The licensee does not consider failure of operators to depressurize as a vulnerability because this action has been emphasized in training and has been judged unlikely to induce operator error.

REFERENCES

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APPENDIX B

HUMAN RELIABILITY ANALYSIS TECHNICAL EVALUATION REPORT