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Kewaunee

Technical Evaluation Report on the Individual Plant Examination Front End Analysis

NRC-04-91-066, Task 16

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1. INTRODUCTION

This introduction section presents the process used by Science and Engineering Associates (SEA) to review the front end portion of the Wisconsin Public Service Corporation (WPSC) Individual Plant Examination (IPE) Submittal for the Kewaunee Nuclear Power Plant. This front end review focuses on accident sequences leading to core damage, due to internal initiating events and internal flooding. Reviews of the human factors and back end aspects of the Kewaunee IPE were performed by the Nuclear Regulatory Commission (NRC) with contractual assistance from Concord Associates, Inc., and Scientech, Inc., respectively.

I.1 SEA Review Process

This report is based on a review of the Submittal and on licensee responses to questions posed to the licensee by the NRC. The licensee responses to these questions are also reflected in this report.

I.1.1 Review of FSAR and Tech Specs

The NRC provided the Kewaunee IPE Submittal to SEA in April, 1993. SEA began work on the Kewaunee review in late May, 1993.

Selected portions of the latest (updated) Final Safety Analysis Report (UFSAR) and Technical Specifications (Tech Specs) for Kewaunee were copied and made available to SEA's lead analyst. The UFSAR copies were made from up-to-date documentation provided by the NRR Project Manager. The Tech Spec copy was obtained through NRC's controlled documentation sources. These UFSAR and Tech Spec copies were reviewed as needed during the course of the overall review.

I.1.2 Review of IPE Submittal

A review of the IPE Submittal for Kewaunee was accomplished. The effort incorporated a review of the technical areas called for the Statement of Work (SOW) for this task and NUREG-1335.

I.2 Kewaunee IPE Methodology

The Kewaunee IPE uses the standard small event tree/large fault tree methodology to perform the Level 1 analyses. WPSC was assisted in the Level 1 analysis by Westinghouse Electric Corp. Functional event trees were developed to define the possible accident scenarios for each specific initiating event. Fault trees were created for each front-line system identified in the logic of the associated event tree(s). Fault trees were also developed for the front-line system support systems, including all of the actuation systems associated with reactor protection and engineered safety features. Both generic and Kewaunee plant-specific data were used to determine the initiating event frequencies and equipment failure probabilities. Recovery actions are considered. Common mode failures are stated to have been incorporated into the fault tree models. Sensitivity studies were performed to determine the impact in core damage frequency from selected factors. Importance analyses were performed to identify important contributors to core damage frequency.

I.3 Kewaunee Plent

The Kewaunee plant is a Westinghouse Electric Corporation designed Pressurized Water Reactor (PWR) employing a 2-loop nuclear steam supply system (NSSS). The rated thermal power is 1650 MWt, and the nominal power output rating is 535 MWe. The Kewaunee design features a dry, low leakage cylindrical steel primary containment structure surrounded by a medium leakage reinforced concrete shield structure.

The Kewaunee Nuclear Power Plant is located in Kewaunee County, Wisconsin, along Lake Michigan's western shoreline. It is located about 30 miles from Green Bay, Wisconsin. The Nuclear Steam System Supplier was Westinghouse Electric Corporation, and Pioneer Service and Engineering (PS&E) was the Architect Engineer (AE). This is the only U.S. nuclear plant for which PS&E provided these AE services. The Kewaunee project was granted a construction permit in 1969, and the unit achieved commercial operation in 1974.

I.3.1 Similar Plants and PSA's

The following plants are Westinghouse 2-loop designs [NUREG/CR-5640]:

Ginna
Point Beach Units 1 & 2
Prairie Island Units 1 & 2

I.3.2 Unique Features

The Kewaunee Submittal identified several "important" design features during the IPE evaluation process. The features of interest to the Level 1 evaluations are as follows:

Level 1 Important Safety Features

- High head safety injection pumps inject at 2200 psig, which is higher than typical Westinghouse plants designated as low pressure plants. This feature allows for early injection to the reactor for LOCA events where the reactor system pressure remains high, and can contribute to a reduced core damage frequency (CDF).
- Three auxiliary Feedwater (AFW) pumps (two motor-driven and one turbine driven for diversity), which are independent of cooling water systems. The service water system serves as a backup suction supply to the three AFW pumps. These features improve the reliability of AFW, and thus tend to reduce CDF.

- Separate eight hour batteries for safeguards and non-safeguards equipment. The eight hour battery life gives more time for recovery of normal power systems compared to plants with lesser battery capabilities, and reduces the likelihood of core damage due to loss of power events.
- Four safety related service water pumps for a single unit site. This feature provides redundant service water capacity, thus reducing the CDF because of the reduced likelihood of incurring a loss of service water.
- The chemical volume and control system has three positive displacement charging pumps, which are independent of cooling water systems. One of the pumps is driven by a variable speed DC motor for speed control and is not dependent on instrument air for attaining maximum pump output. These features improve the reliability of this system, better assuring the availability of this system for injection to the reactor vessel and reducing the likelihood of core damage for certain classes of accidents.
- Two independent methods for maintaining reactor coolant pump seal integrity, seal injection from the charging pumps, and thermal barrier cooling via the component cooling water system. This redundant means of providing seal cooling reduces the likelihood of seal LOCA events, and reduces the frequency of core damage due to seal LOCAs.

Level 2 Unique Safety Features

The Kewaunee containment has large and redundant heat removal capabilities. The containment is equipped with four fan cooling units (FCUs) and two internal containment spray (ICS) trains, each with heat exchangers. The Submittal states that only one FCU or ICS is needed to prevent containment failure on overpressure.

The Submittal states that the containment free volume is such that complete oxidation of the fuel cladding does not produce enough hydrogen to challenge the containment structure. The design also minimizes the likelihood of hydrogen pocketing.

The geometry of the cavity and instrument tunnel is stated to be such that direct containment heating should be minimal following high pressure melt ejections. The Submittal also states that the cavity floor area is large enough to allow the debris to spread into a thin, coolable layer, minimizing the likelihood of non-volatile fission product release.

II. REVIEW FINDINGS

II.1.1 General Overview of Front-End Analysis

II.1.1.1 Completeness of Submittal

The Submittal is complete with respect to the type of information requested in NUREG-1335.

II.1.1.2 Description and Justification for Methodology Used

The Kewaunee IPE Submittal indicated that the methodology followed was similar to that of NUREG/CR-2300. This IPE used the small event tree/large fault tree method. Support system event trees were not used. Rather, support system fault trees were linked to front-line fault trees where appropriate. The methodology used is summarized in Section 2.3 of the Submittal, while additional information is provided in appropriate locations in the documentation.

In the identification of initiating events, the Kewaunee Submittal indicates that other PRAs for similar plants were reviewed, as were PWR generic experience data. Kewaunee-specific operating experience from the fifteen years of plant operation was also reviewed to identify initiating events.

Plant-specific event trees were developed for each of the initiating events selected for evaluation. Important post-initiator operator actions were included in the event trees. System success and failure criteria were developed for the various accident sequences. WPSC indicated that the success criteria were based both on USAR analysis/assumptions and on MAAP analyses performed in support of the IPE, but does not indicate which success criteria were derived from which source (RAI, Responses). System interdependencies were modeled in the fault trees. Systemic event trees were used, and the tree top events represent system responses and operator actions.

The dependencies among systems were noted in the documentation and appear to have been carefully considered. Table 3.2-3 of the Submittal presents the plant system dependency matrix. However, this table is not annotated or footnoted to clarify the nature and extent of the dependencies indicated. Tables 3.2-4 through 3.2-15 indicate the dependence of key components on electrical power (buses needed), coolant, and/or air. These tables also indicate the normal and emergency status of each component listed, i.e., standby, start/run, open, closed, etc. The extent of dependence on support systems is not clear. That is, whether partial capacity of support systems or functions was taken credit for in the analysis, or whether support credit was taken only when full capacity was available.

System fault trees were not included in the Submittal. However, for each front-line and support system all associated fault trees developed to model system performance are explicitly listed and are very briefly discussed. The fault trees were stated to be developed down to the component level. Models of the feedwater and instrument air systems were also developed. The fault tree logic models are stated to have been quantified using the Westinghouse GRAFTER2 and SIMON5 software.

In summary, the methods used to perform the Kewaunee IPE are clearly described and are comparable to the methods used in other IPEs. The methodology employed is the small event tree/large fault tree approach.

II.1.1.3 Assurance of Use of As-Built, As-Operated Plant

Section 2.4.3 of the Submittal lists the "PRA Basis Documentation" used in developing the Kewaunee IPE, and includes documents/information such as the Updated Final Safety Analysis Report (UFSAR), Technical Specifications, Operating Procedures, system drawings and descriptions, plant walk-throughs, maintenance records and procedures, and design change packages. This section also states "the Kewaunee PRA has been kept up-to-date with plant modifications and procedural changes in order to maintain it as a 'Living PRA'." The licensee defined a living PRA as one which can be used as a tool in decision making for the life of the plant (Submittal, p. 1). WPSC stated that the modeling used in the IPE reflected plant design and operational procedures as they existed as of December 1, 1992 (RAI, Responses). Their analysis used failure rate data and test and maintenance unavailabilities current as of December 1989.

The Kewaunee IPE Submittal states that certain plant improvements that have not yet been implemented have been taken credit for in the PRA. All such improvements were stated to have been approved by the plant management and "are scheduled for completion in the near future" (Section 6.3).

The Submittal states that WPSC personnel performing the analysis included a Senior Reactor Operator (SRO) and a Shift Technical Advisor (STA) with knowledge of the plant and with ready access to plant systems to check system configurations. The IPE Submittal states that official plant walkdowns with the PRA contractor staff were conducted for the internal flooding and Level 2 evaluations. Informal walkdowns were also stated to be performed whenever there was any doubt of a system configuration. Initially WPSC personnel performed about 50% of the IPE. However, the Submittal states that since September of 1991 WPSC staff performed about 95% of all Level 1 activities.

In summary, the IPE Submittal and subsequent licensee responses to NRC questions indicates that the analysis performed reflected the plant design as of December 1, 1992. Plant walkdowns and other methods were employed to help assure that the models used reflected the actual plant design. In addition, WPSC staff with Kewaunee plant experience were assigned to assist in the IPE. However, the documentation indicates that this was not done for all systems, but rather only when particular system configuration questions arose. The models employed took credit for some physical modifications and procedural changes which have not yet been implemented.

II.1.1.4 Internal Flooding Methodology

The internal flooding methodology is described in Section 3.3.8 of the Data Summary Sheets. That section identified the specific tasks or steps that were used to evaluate flooding hazards and effects, and it reviews the specific assumptions and bases that were used. Existing flooding studies for Kewaunee were first reviewed, and flooding events that could result in a plant trip or endanger safe shutdown components were identified. Sixteen areas with possible flooding vulnerabilities were identified, and walkdowns of these areas were performed to further evaluate equipment in each area that could be affected by flooding. Based on these walkdown and preliminary evaluation efforts, six areas were found to warrant more intensive analysis. The Submittal describes the analysis performed for each of these six areas, and includes a listing of the equipment in each area that could be damaged by flooding. The IPE Submittal review indicated that the Kewaunee flooding analysis was generally complete, with the exception of the effects of sprays.

The Submittal and other information provided by WPSC indicate that spray-induced damage to safety equipment was considered for items within a 10 ft. radius of possible break/leak locations. However, it is not clear from this information if sprays were considered to be credible for insulated low-energy water piping, or if leaks in these systems were treated only as drip sources. The limitation for affected equipment to a 10 ft. radius from a leak or break would appear to be optimistic since water jets from even medium energy piping are likely to effectively carry considerably farther than 10 feet. The information provided by the licensee is not sufficient to assess the importance of this particular assumption or the impact on core damage frequency of using a less optimistic basis for spray-induced failures. The contribution to CDF from internal flooding events, per the Kewaunee Submittal, is less than 1%.

II.1.1.5 Peer Review

Section 5.2 of the Kewaunee IPE Submittal describes the project review process followed by WPSC. Reviews were performed both by an independent internal group and by an independent external review team. The independent internal group apparently had little prior PRA experience, but they did have knowledge of the Kewaunee plant systems. This group's review was sufficient such that they provided recommendations regarding system modeling. WPSC also enlisted the assistance of a team of six experts from other companies to review each of the PRA tasks and the methodology used. These outside experts were from Battelle, Safety Management, Inc., Sargent & Lundy, and Wisconsin Electric Power Company. This external team focused on methodologies and overall project quality, and in-depth reviews were performed on selected portions of each area. This team also focused on those areas that WPSC staff did not have a great deal of experience (stated to be Level 2 PRA, containment analysis, human reliability analysis, and common cause).

In addition to the above reviews, a member of WPSC's engineering staff served on the Point Beach independent review team. That participation resulted in several suggested improvements for the Kewaunee IPE.

Based on the descriptions provided, it is concluded that the Kewaunee IPE models and analysis were subjected to independent reviews by both plant staff familiar with the plant systems and plant operation, and by an external team with expertise in PRA methodology (Level 1 and Level 2). These reviews resulted in improvements to the Kewaunee PRA, and helped assure that the PRA techniques, data, and methodology were correctly applied.

II.1.2 Review of Accident Sequence Delineation and System Analysis

II.1.2.1 Identification of Initiating Events and Related Dependencies

Section 3.1.1 of the IPE Submittal states that initiating events were identified by reviewing NUREG/CR-3682 (IPE Submittal Ref. 17) and past PRAs, and by performing a system-by-system review to determine which system failures can cause a plant trip. These examinations covered plant trip events that can or have occurred at Kewaunee that are not included in the list of generic initiating events. A list of past PRAs that were reviewed was presented. However, the Submittal did not provide a systematic presentation of the system reviews with regard to initiating events.

The Submittal states that a "Core Damage Logic Diagram for Internal Initiators" was developed to systematically categorize all internal initiators on the basis of similar transient progression or consequences, but does not provide any further information about this process or the resultant logic diagram.

Section 3.1.1 of the Submittal further states that initiators were grouped based on plant response, signal actuation, systems required for mitigation, and subsequent plant-related effects. There is some clarification of the definition of each initiator group.

The initiating events are consistent with the NRC-sponsored Sequoyah PRA (NUREG/CR-4550, Vol. 5). Very small loss of coolant accidents (LOCAs) can be maintained by normal charging pump flow and do not result in a reactor trip; therefore, they were not modeled. The Submittal indicated that initiating events were grouped into three categories; LOCAs, transients, and special initiating events. The IPE treats station blackout, feedwater/steam line breaks, and anticipated transients without scram (ATWS) as separate initiators.

Generic data were used to establish initiating event frequencies for events where no Kewaunee-specific experience was available. Frequencies for large and medium LOCAs, steam generator tube rupture, interfacing system LOCA, station blackout, loss of offsite power, and ATWS were based on generic data. The frequencies for transients with and without feedwater were based on plant-specific data. For loss of service water, loss of component cooling water, loss of emergency electrical buses, and loss of instrument air, the initiating event frequencies were established by quantification of the system fault-trees. The Kewaunee IPE omitted the loss of a 4160 VAC bus and other electrical buses as initiators. However, the licensee performed the analyses of loss of 4160 VAC buses, both vital and non-vital, and included these initiators in a revised analysis (RAI, Responses).

The licensee provided a list of mitigating functions required for each initiator and provided dependency tables that identify dependencies between systems, but there is no table that identifies dependencies between initiating events that are not support system failures and their mitigating functions. The treatment of such dependencies in the development of accident sequences is discussed under II.1.2.5.

The list of internal initiating events was reviewed to ascertain which events could be induced by a flood at Kewaunee. For initial screening, a maximum flood that disables everything in that room was postulated. If a reactor trip could be initiated by the maximum flood, the room was further studied with calculations and a walkdown. Six flooding scenarios that were found to warrant further investigation became the six internal flood initiating events.

Thus, the licensee has described the process used to identify initiators (including internal flood) and has taken into account both generic and plant-specific information. Furthermore, the initiating events considered and their frequencies appear to be consistent with those considered in other PRAs.

II.1.2.2 Identification and Analysis of Front-Line and Support Systems Important to the Prevention of Core Damage and Mitigation of Fission Product Release

The Kewaunee IPE Submittal states that Level 1 front-line systems are those that are used to maintain reactivity control, reactor coolant system inventory, and reactor coolant system heat removal capability. Support systems are those necessary for the successful operation of the front-line systems, either directly or through the support of other support systems. The systems modeled or considered in the IPE are listed below:

- Auxiliary Feedwater System (AFW)
- Component Cooling Water (CCS)
- Containment Air Cooling (CAC)
- Containment Isolation (CI)
- Electrical Power (4160/480/120V AC, 120V DC)
- High Pressure Safety Injection (HPSI)
- Internal Containment spray (ICS)
- Low Pressure Safety Injection (LPCS)
- Main Feedwater (including condensate) (MFW)
- Reactor Protection (including reactor trip system and emergency safety feature actuation system (ESFAS)) (RPS)
- Service Water (SW)
- Miscellaneous Systems, including:
  - Chemical & Volume Control
  - Reactor Coolant Charging
  - Station and Instrument Air

The Kewaunee IPE Submittal described each of the foregoing systems. The discussions for each system presented the following information:

- System function,
- System description, and
- Listing of fault trees developed to model the system for particular initiating events and accident sequences.

Front-line and support systems drawings are provided.

The associated fault tree descriptions (listings) included brief statements of important success criteria used in the modeling.

Heating, ventilating, and air conditioning (HVAC) requirements critical for the operation of front-line systems are discussed with the pertinent accident scenarios and/or system descriptions. HVAC dependencies are indicated in the dependency matrix (Table 3.2-3) and in the component dependency tables (Tables 3.2-4 through 3.2-15). The Submittal does not indicate whether or not HVAC is needed for successful control room operation. The licensee provided additional analysis of the impacts of HVAC. This reanalysis provided additional insights as to the importance of HVAC, and indicated the relative changes in CDF due to the loss of HVAC to particular areas of the plant.

The Kewaunee IPE Submittal indicates that the preparers performed a review of the frontline and support systems important to the prevention of core damage and mitigation of fission product releases.

II.1.2.3 Treatment of Dependencies (Including Asymmetries) Among Plant Systems; Dependency Matrices

Section 3.2.3 of the Kewaunee IPE Submittal specifically addresses dependencies. As noted in Section II.1.1.2 above, the dependencies among systems were documented and were considered. Table 3.2-3 of the Submittal presents the plant system dependency matrix. However, this table is not annotated or footnoted to clarify the nature and extent of the dependencies indicated. Tables 3.2-4 through 3.2-15 indicate the dependence of key components of front-line and support systems on electrical power (buses needed), coolant, and/or air. These tables also indicate the normal and emergency status (upon receipt of ESF actuation signal) of each component listed, i.e., standby, start/run, open, closed, etc.

For the high and low pressure injection systems, the dependencies are listed separately for the injection and recirculation phases of emergency core cooling system (ECCS) operation. The reliance on electric power, ESF actuation, instrument air, service water, the component cooling water systems, and air cooling (HVAC) is explicitly noted in Tables 3.2-4 through 3.2-15. The systems whose major components are included in these tables are:

- Low pressure injection (both injection and recirculation modes)
- High pressure injection(both injection and recirculation modes)
- Service water
- Component cooling water
- Auxiliary feedwater
- Main feedwater
- Residual heat removal (RHR)
- Chemical and volume control
- Containment spray
- Instrument air (IA)
- Containment air cooling

Dependency information is also presented in the individual system descriptions (Section 3.2.1), but such information is not uniform from one system description to the another. Also, many of the components listed in Tables 3.2-4 through 3.2-15, particularly for the service water system, could not be found in the simplified system schematics provided in the Submittal.

HVAC requirements were evaluated in the IPE. All safety systems potentially dependent on the availability of HVAC were identified in the dependency matrix (Table 3.2-3) and in related tables (Tables 3.2-4 through 3.2-15). The discussions in the Submittal do not indicate that any evaluations were performed to determine if safety system equipment could function satisfactorily in the adverse conditions without the HVAC system. Also, although dependence on HVAC is indicated in Table 3.2-3, the nature of that dependence for any particular system is not discussed in any consistent manner from one system to another. Table 3.2-3 appears to be somewhat incomplete in that for at least one system (component cooling water) the dependence on HVAC is discussed in the system description but is not noted in Table 3.2-3. Finally, the portions of particular systems which fail on loss of HVAC (electric motors, motor control centers, electrical cabinets, etc.) are not identified. The Submittal does not indicate whether or not HVAC is needed for successful control room operation.

The licensee performed additional analyses to check the validity of HVAC assumptions used in the IPE analyses. Based on the new analysis, the original IPE models for affected systems/areas were either verified as being sufficient or were modified to reflect the newer insights. The licensee provided new information as to the relative importance of HVAC failures on CDF. However, the absolute changes in CDF are not cited in the additional information provided by the licensee, nor is a new estimate of overall CDF.

Overall, the Kewaunee IPE Submittal indicates that system and component dependencies were considered, and there are no apparent errors or omissions that can be detected from the information provided.

Asymmetries within systems in the Kewaunee plant are not specifically discussed. However, they appear to be accounted for in the evaluation process.

II.1.2.4 Treatment of Common Cause Failures

The IPE used the Multiple Greek Letter (MGL) method to quantify common cause failures (CCF) within a system. The Submittal contains a brief tutorial on application of the MGL methodology.

The Submittal provides a list of component groups for which common cause events "may be defined" and a shorter list of component groups for which parameters were assigned. The common cause failures for other component groups were omitted. The licensee indicated that the criteria used to eliminate certain common cause groups from further consideration was that, for components that were often exercised through daily use or surveillance testing, plant experience showed no CCF for those component groups. This criteria was applied to electrical equipment such as batteries and most breakers. In most cases, therefore, CCF of electrical equipment was not modeled in the Kewaunee IPE unless system-specific conditions warranted it.

Section 3.3.4 of the IPE Submittal contains the sentence, "A check of the plant specific data for common cause events is made prior to using the generic factors, thereby ensuring the generic factors are conservative." The licensee stated that the plant-specific data analysis performed did not identify any additional common cause events or precursors that would suggest that other-than-generic factors should be used. Thus, only generic factors were used. Most of the MGL factor values were calculated using data from EPRI NP-3967, other factor values were based on data from NUREG/CR-4780. The following table compares the Kewaunee MGL beta factors used for particular components with those derived from ~~the NUREG/CR-4550~~ methodology.

Component	Kewaunee β (beta)	NUREG/CR-4550 β (beta)
Diesel Generators	0.025	0.038
Motor Operated Valves	0.038	0.088
Pumps		
high head	0.10	0.21
residual heat removal	0.077	0.15
containment building spray	0.057	0.11
auxiliary feedwater	0.021	0.056
service water and CCW	0.032	0.026

The table indicates that the Kewaunee CCF values are typically about a factor of two lower than those derived from the NUREG/CR-4550 source. The values used are well within the overall uncertainty ranges applicable to common cause factors.

In summary, the IPE used the MGL method to model common-cause failures within systems. Only generic data were used since a review of plant-specific data did not identify any CCF. The CCF factor values used are generally consistent with those used in other PRAs. The Kewaunee IPE approach for dealing with CCF is somewhat simplified and optimistic in that certain component groups such as batteries and some breakers were not treated as being susceptible to common cause failures.

II.1.2.5 System Event Trees and Special Event Trees; Treatment of Initiating Events, Associated Success Criteria, and Dependencies Between Top Events

We reviewed the identification and quantification of initiating events in the IPE. Initiating events were grouped into three categories: LOCAs, general transients, and plant specific initiating events. The IPE used NUREG/CR-3862 and other PRAs to assist in the identification of transient initiating events. The frequency for a large LOCA was taken from NUREG/CR-4550; the frequency for a medium LOCA was taken from WASH 1400 combined with a calculated value for opening of a pressurizer safety valve. The frequency for a small LOCA was taken from WASH 1400 combined with generic Westinghouse data for a seal LOCA. The frequency for a steam generator tube rupture was taken from Westinghouse data. The frequency of an interfacing systems LOCA was calculated specifically for Kewaunee. Plant specific operating data was used as appropriate to quantify general transient initiating events, such as reactor trip with main feedwater available. The methodology from NUMARC 87-00 was used to quantify the frequency for loss of offsite power (LOSP). A review of plant systems was performed to identify plant specific initiating events; plant specific support systems initiating events were quantified using system fault trees.

Table 3.1-1 of the Submittal lists the initiating events that were retained for modeling, and provides the point estimate frequencies for these initiating events. The frequencies of the initiating events used in the Kewaunee IPE are comparable to those events used in other IPE/PRAs.

Sixteen initiating events were analyzed for core damage: 6 LOCAs, 6 general transients, and 4 plant specific transient events. The 4 plant specific transient initiating events were: loss of service water, loss of component cooling water, loss of a 125 V DC bus, loss of a 4160 V AC bus, and loss of instrument air.

Regarding the loss of HVAC, the licensee stated that HVAC systems for five areas have the potential for being considered as initiating events, these being: battery rooms HVAC, control room/relay room HVAC, control room drive equipment room HVAC, screenhouse HVAC, and containment HVAC. The licensee concluded that operator rounds, procedures, and contingency actions provide the basis for excluding failures of these systems from consideration as initiating events.

The following events were also evaluated as potential initiating events: loss of a non 1E 4160 V AC bus and loss of 120 V AC vital power. This evaluation indicated that the effect on CDF from loss of a non 1E 4160 V AC bus has a small impact on overall

CDF, contributing at most only 3.3E-8/year. Due to this small impact, the IPE does not consider loss of a non 1E 4160 V AC bus to be an initiating event. Regarding the loss of 120 V AC vital power, the licensee stated that none of the equipment powered from this source is very important for mitigation of an accident. The licensee concluded that it is not necessary to include loss of 120 V AC vital power as an initiating event.

The success criteria for prevention of core damage were originally based on UFSAR assumptions, with additional refinements made from MAAP analyses. Core damage was defined for the purposes of the IPE, as occurring if the core hot spot temperature exceeded 1200 °F. The licensee indicates that momentary excursions of core temperature up to 1700 °F could occur without core damage, but none of these excursions were observed in any of the MAAP runs.

The event trees that were produced and analyzed were for the most part systemic event trees. Sixteen event trees were provided in the Submittal, one for each of the sixteen initiating events retained for analysis in the original IPE.

In general, we found the event trees to reflect the success criteria and model accident sequences. The following discussions summarize notable aspects of the event.

The model for steam/feedwater breaks does not address isolation of steam/feedwater after the break. [IPE Submittal, Page 90] However, the licensee stated that the IPE model assumed that one SG is isolated, and the CDF for blowdown of both SGs is 3.1E-10/year which is sufficiently small to provide the basis for this assumption.

The model for steam line breaks assumes that boration of the primary is not required to mitigate the accident. This assumption is supported by a calculation performed by the utility that is summarized in the Submittal.

The non-recovery factors used for offsite power are comparable with values used in other IPE/PRAs. The following non-recovery probabilities were used: [IPE Submittal, Page 147] 0.265 by 2 hours, 0.041 by 9 hours, and 0.02 by 11 hours.

The success criteria for ATWS consider the need to trip the turbine. Event "AMS" models the system that initiates turbine trip and actuation of AFW during an ATWS. [IPE Submittal, Page 111] Closure of the turbine stop valves was not modeled, but the licensee stated that if this were modeled the increase in CDF would be negligible.

The success criteria for ATWS indicate that an ATWS event can be mitigated early in cycle life when the moderator temperature coefficient provides the smallest negative reactivity due to high boration. [IPE Submittal, Page 112] The license stated that during the first 40 days of a fuel cycle, both PORVs are required to relieve pressure following an ATWS, but that the IPE assumed that only one PORV was required; consideration of the requirement for both PORVs during the first 40 days increases the CDF by 5E-9/year, a small amount. (It should be noted that at some larger, higher-power PWRs, there is a certain time early in core life for which successful operation of all PORVs and safety valves is insufficient to relieve pressure during an ATWS.)

The Westinghouse seal LOCA model was used in the Kewaunee IPE. This model provides the following probabilities for core uncovery due to a seal LOCA considering recovery of offsite power: 2.83E-2 at 2 hours (with or without cooldown), 7.62E-2 at 9 hours (without cooldown) and 7.07E-2 at 11 hours (with cooldown). This model considers a spectrum of seal leak rates from 21 gpm up to a maximum of 480 gpm per pump. The IPE assumed use of conventional O rings in the reactor coolant pumps (RCPs); in April 1993 the plant replaced the conventional O rings with high temperature O rings, thus making the IPE seal LOCA model less optimistic for the current components in service. Numerous IPEs have used the Westinghouse seal LOCA model, and in general these IPEs conclude that seal LOCAs are less of a contributor to overall CDF than IPEs that use a less optimistic model for seal LOCAs such as used in NUREG/CR 4550. Appendix A of this report summarizes the calculations used to compare core uncovery probability estimates vs time for various seal LOCA models. Figures A-3 and A-4 of this appendix compare data for the seal LOCA models used in the Kewaunee IPE, the Point Beach IPE, and the NUREG/CR 4550 PRA for Surry.

Based on the tabulation of dominant sequences in Table 3.4.4-4, seal LOCAs as a result of loss of seal cooling contribute about 6% to the overall CDF in the IPE Submittal (sum of sequences #6, #25, and #39). RCP seal LOCAs overall contribute 5.6E-6/year to CDF, about 8% of the total CDF, in the IPE model; in a revised model of the Kewaunee IPE, RCP seal LOCAs overall contribute 1.8E-5/year, with the increase largely due to a higher human error probability assigned to establishing seal cooling using the Technical Support Center diesel generator (TSC DG). The licensee also states that the ability to provide seal cooling during station blackout using a charging pump powered off the TSC DG has an important effect in lowering the contribution of seal LOCAS to overall CDF. Without this option, the response states that the overall CDF reported in the Submittal would increase by about 47%.

Regarding the condensate storage tank (CST) inventory available to support core cooling during a station blackout, the tech specs require a minimum CST inventory of 39,000 gallons which would supply adequate inventory for 4 hours. However, the two CSTs are maintained filled with a total inventory of 150,000 gallons, which is sufficient for about 16 hours.

The data used to quantify the probability that low pressure piping exposed to beyond design basis pressure does not fail was taken from NUREG/CR-5102. This data was used in modeling the interfacing systems LOCAs.

In the revised IPE model, the following HVAC dependencies have been added: HVAC for the diesel generators, HVAC for the control room/relay room and the battery rooms, and HVAC for the motor driven AFW pumps. The IPE was updated to reflect these additional support system dependencies. However, the information provided does not identify the change in CDF due to these changes in the model.

II.1.2.6 Identification of Most Probable Core Damage Sequences and Dominant Contributors; Consistency with Insights from PSAs of Similar Design

The estimated core damage frequency for Kewaunee, as reported in the Submittal, was calculated to be 6.65E-5/yr. The dominant accident sequences listed in the Kewaunee IPE Submittal are presented in Section 1.4, "Summary of Major Findings" and are discussed in more detail in Section 3.4. A listing of the top 71 CDF sequences is given in Section 3.4, "Results and Screening Process." That section lists the Generic Letter 88-20 screening criteria used in the Kewaunee evaluation. The Section 3.4 discussions review the top 13 accident sequence types which contribute about 85% to the total core damage frequency. For each sequence the initiator is identified together with the initiating event frequency and its contribution to core melt frequency. Each of the 13 sequences is briefly described in terms of event successes and failures of specific systems, components, and operator actions. However, the specific causes of failures are not discussed for all sequences. The primary reasons for core melt are identified, as are specific modeling assumptions. The failures of systems and components included failures of operators to take the proper action in the time available. A statement is also made as to whether the sequence is plant specific or is typical of many PWRs. Of the 13 sequences discussed, only the loss of instrument air is stated to be plant specific because of the design and reliability of this system.

The following table identifies the top eight contributors to core melt referenced by initiating event as presented in the Kewaunee IPE Submittal. For comparison, the Point Beach Nuclear Plant IPE results indicated a total CDF of 1.15E-4/yr., and NUREG/CR-4550 for Surry indicates a mean CDF of 7.4E-5/yr. Point Beach is a Westinghouse 2-loop design similar to Kewaunee; Surry is a Westinghouse 3-loop design. The overall CDFs for Kewaunee and Point Beach are within a factor of two of each other, and the Kewaunee CDF is in close agreement with that for Surry. Note that station blackout is not a true initiator, but represents loss of off-site power and various transients coupled with failure of the plant emergency diesel generators.

Rank	Initiating Event	Kewaunee CDF Frequency/%
1	Station Blackout	2.6E-5/40%
2	Small LOCA	1.4E-5/21%
3	Medium LOCA	8.1E-6/13%
4	SG Tube Rupture	5.3E-6/8%
5	Loss of Offsite Power	4.5E-6/7%
6	Transient with MFW	2.7E-6/4%
7	Loss of Instrument Air	2.1E-6/3%
8	Large LOCA	1.9E-6/3%
9	Other	6.7E-7/1%
	Overall CDF	**6.65E-5**

The dominant Cut Sets for each Kewaunee dominant IE are as follows:

Station Blackout:	failure of turbine driven (TD) AFW, power not restored within 2 hours
Small LOCA:	failure due to operator error and failure to establish containment sump recirculation
Medium LOCA:	failure of ECCS due to failure of long term recirculation
Steam Generator Tube Rupture (SGTR):	failure to cooldown and depressurize reactor cooling system (RCS).

In summary, the Kewaunee IPE identified the most probable core damage sequences and identified the dominant contributors to each sequence. Comparisons with PRA results developed for other PWRs were not presented. However, our comparisons with other PRA results indicate that the Kewaunee CDF estimates are generally consistent with results developed for other PWRs.

II.1.2.7 Front-End and Back-End Dependencies

The IPE assumes that CCW cooling of one RHR heat exchanger is sufficient to provide heat removal from containment to support core cooling.

The UFSAR indicates that the net positive suction head required (NPSHR) for the RHR pumps is 14 feet at runout flow, and that the drop in elevation from the top of the containment sump to the location of the RHR pumps is about 24 feet. [UFSAR, Table 6.2-6 and Figure 1.2-8] Assuming a typical pressure drop for flow losses, it appears that the RHR pumps can operate with a saturated sump.

We performed scoping calculations to determine whether or not overheating of the RHR pumps might occur while using only one RHR heat exchanger for heat removal. Our calculations summarized in Appendix A of this report indicate that over-temperature is not expected to interrupt core cooling when only one RHR heat exchanger is used.

Section 3.1.5.1 of the Submittal discusses the process used to bin core damage sequences into plant damage states (PDS) for back-end analysis. The following parameters were used for the binning: status of the low pressure recirculation system, status of the containment air cooling system, status of the containment spray system, status of containment isolation, timing of core melt, and the RCS pressure at the time of core damage. These parameters for binning into PDSs are comparable to those used in other IPE/PRAs.

II.1.2.8 Consideration of Initiating Events Affecting More Than One Unit; Treatment of Systems Shared Between Units

Kewaunee is a single unit at a one unit site. It does not share any systems with any other units.

II.1.3 Review of the IPE's Quantitative Process

II.1.3.1 Quantitative Evaluation of the Impact of Integrated System and Component Failures on Plant Safety; Use of Mean Values and Sensitivity Studies

The Kewaunee IPE quantitatively evaluated the impact of integrated system and component failures on plant safety. The quantification process used to estimate unavailabilities of systems and functions is described in Section 3.3.5/6/7.

The IPE Submittal states that fault trees were developed to model systems represented in the event tree top events. The system descriptions provided in Section 3.2 included listings of the related fault trees used in the Kewaunee IPE Level 1 evaluations.

Sensitivity and importance analyses were performed.

Sensitivity analyses were conducted on initiating event frequencies, operator actions, risk modeling, and plant design. Dominant initiating event frequencies were changed by reviewing system configurations, other data bases, possible alternatives that would prevent frequent occurrence of the initiating event, and the like, to determine a range of values for the frequency so that the variability of the core melt frequency could be assessed. Risk modeling sensitivity evaluations were stated to include changes to system success criteria, analysis assumptions, and other modeling criteria. Design alternative studies included conceptual changes to systems whose failure contributes substantially to core melt frequency. These sensitivity studies address the expected major sources of uncertainty in the evaluation.

The Kewaunee IPE Submittal discussed how each sensitivity evaluation was carried out and gave the results in terms of changes in core melt frequency.

The results of IPE sensitivity evaluations are summarized in the following table. The table presents the parameter of interest, its nominal value and extent of variation for the sensitivity study, and a statement of the insights gained.

Parameter Varied	Nominal Value	Sensitivity Range	Insights Gained
Cut-off probabilities during quantification for reduction of fault tree cut set files	1.0E-9, reduction of fault tree cut set files for LSP sequences 1.0E-12, reduction of fault tree cut sets for other sequences 1.0E-9, for LSP sequence linking 1.0E-10 for other sequence linking	1.0E-12, reduction of fault tree cut set files for all sequences 1.0E-10, for LSP sequence linking 1.0E-11, for other sequence linking	Additional 10,000 cut sets obtained; total plant CDF increased by only 1.6%. Therefore, base case cutoff value is appropriate.
Multiplier on all operator action failure probabilities	1.0	0.2 to 5.0	If all operator actions successful, CDF improved by 25%; factor of five increase in human error rates increases CDF by factor of 3
Multiplier on all common cause probabilities	1.0	0.0 to 5.0	Elimination of CCF improves CDF by 25%; fivefold increase in CCF increased CDF by factor of 2.
Multiplier on failure probabilities for all air-operated valves (AOVs)	1.0	2.0	Increase in AOV failure rate by factor of 2 increased CDF by 10%.
Multiplier on failure probabilities for all motor-operated valves (MOVs)	1.0	2.0	Increase in MOV failure rate by factor of 2 increased CDF by 24%.
Multiplier on failure probabilities for all diesel generators	1.0	2.0	Increase in DG failure rate by factor of 2 increased CDF by 24%.
Multiplier on all test and maintenance unavailabilities	1.0	5.0	Factor of 5 increase in test and maintenance unavailabilities increased CDF by 12%
Multiplier on loss of instrument air (INA) initiating event frequency	1.0	10.0	Increase in INA failure rate by factor of 10 increased CDF by 24%.

The Kewaunee IPE Submittal also included an importance analysis of initiators, core melt sequences, cutsets, and components. The initiator importance results were discussed in II.1.2.6 above. The top 71 core melt sequences are presented in order of contribution to CDF in Table 3.4.4-4 of the Submittal. The top 50 cutsets are presented in order of contribution to CDF in Table 3.4.4-6 and -7 of the Submittal. No particular insights are discussed. The results of the importance by component indicated that the auxiliary feedwater systems contributed about 32% to the total CDF, with reliability of the AFW pumps being the major contributor.

The licensee also performed additional importance and sensitivity analyses. These analyses included an importance assessment of CCF, RCP seal LOCA, loss of injection, loss of decay heat removal (DHR), systems whose failures contribute most to CDF, and component maintenance and test. Common cause failures are stated to contribute 26.3% of the total CDF in the IPE as submitted and 21% in the revised model. The four systems that dominate CCF are low pressure recirculation (11.5%), auxiliary feedwater (2.9%), service water (2.7%), and safety injection signal (0.2%). The discussion further identifies the components of each system that contribute to the CCF of the system. A key insight gained is that the MOVs of the low pressure recirculation contribute the most to CCF. The licensee indicated that as the Kewaunee MOV inspection and test program is implemented, it is expected to reduce the likelihood of valve failures.

The importance of RCP seal LOCAs was discussed in Section II.1.2.5. These failures contribute about 1.8E-5/yr to CDF. Loss of injection and loss of DHR contribute about 2.7E-5/yr and 2.6E-5/yr, respectively. Thus, all contribute about equally.

The systems importance analysis listed the ten systems that contribute the most to CDF. These systems, and their relative contribution to CDF, are as follows:

System	Importance per Revised IPE Model Original IPE & Revised Model
4160 VAC Power	28.7, 33.9%
Residual Heat Removal (incl. LPSI)	28.6, 27.1%
Diesel Generators (incl. TSC Diesel)	24.6, 28.9%
Auxiliary Feedwater	23.7, 21.3%
Service Water	15.3, 14.7%
Station and Instrument Air	5.1, 1.5%
Engineered Safety Features Actuation	4.0, 1.7%
480 VAC Power	3.6, 4.1%
Main Steam and Steam Dump	2.7, 2.5%
Chemical and Volume Control	2.5, 2.7%

An insight gained from this importance analysis is that some non-safeguards systems such as instrument air and main steam and steam dump are among the ten most important systems. Also, six of the top ten systems are support systems. An additional insight is that the auxiliary feedwater system is the major test and maintenance contributor to CDF.

The initiating event frequencies and basic event probabilities were developed as point estimates, intended to represent maximum likelihood estimates, not as probability distributions (IPE Submittal Sec. 3.3.1/2).

In summary, the Kewaunee IPE quantitatively evaluated the impact of integrated system and component failures on plant safety. Sensitivity studies were performed on operator actions (including recovery actions), common cause, test and maintenance, and failure rates for particular system components. The sensitivity to analysis cutoff frequency was also assessed. The information provided also indicated that the licensee gained an appreciation of system importance. The analysis used point or central estimate values to quantify accident sequences.

II.1.3.2 Consistency of Techniques Used to Perform Data Analysis

The Submittal states that each data point in the database was either generic, plant-specific, or a Bayesian update of generic data by plant-specific data. Plant-specific data for fifteen years of operations were pooled for similar components. The Submittal reports that the Bayesian update calculations were done with the BAYES3 code of the GRAFTER Code System.

The IPE Submittal states failure probabilities were based on more recent data in cases where plant improvements or modifications had a positive effect on component availability. Otherwise, the full fifteen years of data were used.

Sources of plant-specific data were:

Maintenance Work Requests
Incident Reports
Licensee Event Reports
Kewaunee Diesel Generator Reliability Study
Kewaunee Nuclear Plant Reliability Data System
Kewaunee Auxiliary Feedwater PRA

The licensee used a generic value from NSAC-182 (IPE Submittal Reference 21) for loss of offsite power events that are plant-centered losses or grid disturbances. These were supplemented by plant-specific, weather-related losses calculated using the methodology of NUMARC 87-00 (IPE Submittal Reference 20). The resulting value, 0.044 per year, is at the low end of the range of values contained in the NREP data base. The NREP values for LOSP range from 0.04 to 0.2 per year, depending on the regional power grid.

Maintenance and test unavailabilities were stated to have been calculated from the frequency of the maintenance activity and the average unavailability duration per such activity. Thus, these unavailabilities are based on plant-specific data rather than generic data.

Information provided by the licensee stated that generic data was used for the containment sump strainers, the technical support center diesel generator, the charging pump discharge relief valves, and the station air compressors. The licensee also identified the components for which plant specific data was used, components for which pooled data was used, and components for which component-specific data was used in a pooled manner.

The techniques used are generally consistent with those used in other PSAs, and plant-specific data are used for nearly all components and systems and for maintenance unavailabilities.

II.1.3.3 Sources of Generic Failure Data and Rationale for Their Use

The licensee identified the sources of generic data for initiating event frequencies as follows:

Initiating Event	Source
Large LOCA	NUREG/CR-4550
Medium LOCA - pipe break	WASH-1400
Medium LOCA - stuck open valve	Plant experience data for transients and generic data for failure to reseat
Small LOCA - pipe break	WASH-1400
Small LOCA - RCP seal failure	Generic Westinghouse experience
Steam Generator Tube Rupture	Generic Westinghouse experience (per tube)
Reactor Vessel Failure	WASH-1400
Large Steam/FW Line Break	5 x Large LOCA

For transient initiators whose frequencies were calculated from fault trees (e.g., loss of service water, loss of component cooling water, loss of 125V emergency DC bus), their "generic data" are those that may have been used to develop component failure probabilities, as described in IPE Submittal Section 3.3.1.

In Section 3.3.1/2 of the Submittal, the licensee identified the primary source of generic data to be NUREG/CR-4550. That data base is consistent with the data contained in NUREG/CR-2815, Appendix G.

Section 3.3.8.1 of the Submittal states that the calculation of flooding initiating event frequency accounted for flood initiation, automatic and human detection, and automatic and human isolation. In several cases, failure data obtained from other PRAs was used as a basis for these frequencies and probabilities, but was modified to account for conditions pertinent to Kewaunee. In other cases, generic data for pipe break frequencies from WASH-1400 was used, but was also adjusted to reflect Kewaunee design and conditions. Section 3.4.4 states that the flooding initiator frequencies used in the IPE are less optimistic than best estimate values. However, as noted in Section I.1.1.4, the Kewaunee IPE assumptions regarding the radius of influence of water sprays appear to be more optimistic than best estimate values. Otherwise, their approach and basis is comparable to that used in other IPEs/PRAs.

We conclude that the licensee has explicitly identified the sources of generic data, including sources used for internal flooding initiators. The IPE generic data are generally consistent with the data contained in NUREG/CR-2815, Appendix C.

II.1.3.4 Common Cause Failure Data and Data Sources

The IPE quantifies the contribution to core damage frequency from common cause failures. The Submittal lists the following common cause failures that appear in the 50 most important cut sets:

Event Identifier	Event Prob.	System or Component	Description In IPE Submittal
34---RHR-----CM	9.8E-4	RHR Recirculation	COMMON CAUSE FAILURE OF RHR
02----SW-----CM	6.2E-5	SW System	NO SERVICE WATER DUE TO COMMON CAUSE FAILURES
05B-SY1FAULT-CM	2.1E-4	AFW System	COMMON CAUSE FAILURES OF AFW SYSTEM
55--SY---SIAB-CM	1.5E-4	ESFAS Train	COMMON MODE FAILURE OF TRAIN A AND TRAIN B
35-OS1OS2EC4-CM	8.7E-5	RCS (Cooldown & Depressurization)	FAILURE DUE TO COMMON CAUSE FAILURES
02-SY-SWIE---CM	1.2E-4	SW System	SERVICE WATER SYSTEM COMMON CAUSE
34I--LPI-----CM	4.2E-4	LPSI Train	COMMON CAUSE FAILURE OF BOTH LPI TRAINS

The licensee identified EPRI NP-3967 and NUREG/CR-4780 as the sources of generic component data for common cause parameters for component groups.

It is not possible to directly relate the text of IPE Submittal Section 3.3.4 and the content of Table 3.3.4-2 with the common cause probabilities in the master data file

(Table 3.1.1-1). The master data file contains probabilities for common cause failures of trains and systems, whereas IPE Submittal Section 3.4 discusses component groups. For some systems Table 3.1.1-1 contains more than one common cause event related to the top events.

Information provided by the licensee indicated that common cause failures contribute about 26% of the total CDF in the IPE as submitted and 21% in the revised model. The four systems that dominate CCF are low pressure recirculation (11.5%), auxiliary feedwater (2.9%), service water (2.7%), and safety injection signal (0.2%). The discussion further identifies the components of each system that contribute to the CCF of the system. The MOVs of the low pressure recirculation contribute the most to CCF. An enhanced MOV inspection and test program is to be implemented to reduce the likelihood of valve failures.

Overall, the licensees treatment of common cause failure data is comparable to that used in other PRAs/IPEs. Sources of data were identified, and the use of CCF data in the IPE was explained.

II.1.4 IPE Approach to Reducing the Probability of Core Damage or Fission Product Release

II.1.4.1 Core Damage Vulnerability and Efforts to Uncover Vulnerabilities; Plant Modifications (or Safety Enhancements) to Eliminate or Reduce the Effect of Vulnerabilities

The Kewaunee IPE defined a vulnerability as "a feature in plant design, procedures, training, etc., which results in a contribution to core melt risk greater than what is expected." (However, nowhere does the document specifically state what is expected in this regard.) Although not specifically stated, the implied method for defining vulnerabilities was to look at those accident sequences that fell within the bounds of the screening criteria stated in Appendix 2 of Generic Letter 88-20.

The search for vulnerabilities focused on the dominant accident sequences. The systems, trains, and components whose failure contributed to these sequences were identified in the discussion of vulnerabilities. Operator actions which contribute to dominant core melt sequences are also called out. Nine specific vulnerabilities are discussed in Section 3.4.3 of the Submittal. The discussion of planned plant improvements indicated actions to be taken on specific components and/or specific procedures. Thus, the Kewaunee IPE went beyond the system level in attempting to assess vulnerabilities.

The particular sequences wherein vulnerabilities were addressed in the Kewaunee IPE included interfacing system LOCA events, internal flooding, loss of offsite power (LSP) and station blackout (SBO).

The plant improvements to Kewaunee that were identified as a result of the IPE have already been taken credit for in the analysis results. The Submittal does not state what quantitative benefit is achieved in reduced core melt frequency by implementing the noted improvements. A few other possible improvements are still under consideration by WPSC. WPSC stated that records were not kept of the CDF before given improvements were credited. Therefore, they have no quantitative estimates of the reduction achieved in CDF due to these improvements.

The approach taken in the Kewaunee IPE to identify plant vulnerabilities provided a means for identifying particular systems or particular operator actions which, if improved, could reduce the potential for core damage events at the plant. However, the overall rationale and criteria for selecting improvements to reduce vulnerabilities was not specifically stated other than that a reduction in CDF was expected.

In summary, the analysis supports the licensee's definition of vulnerability with respect to core damage. The analysis technique included consideration of potential failures down to the component level. The licensee's definition of core damage vulnerability has been used to identify plant vulnerabilities and potential safety enhancements. However, the licensee's analysis is such that the quantitative benefit in CDF reduction from the improvements cannot be estimated.

II.1.4.2 Identification of Plant Improvements and Proposed Modifications Expected to Enhance Plant Safety

The evaluation of insights for plant improvements developed from the IPE process is described in Section 6 of the Kewaunee IPE Submittal. That section discusses planned improvements to the plant configuration and operating procedures that resulted from IPE-developed insights, and it discusses other insights which may be implemented at a later time, depending on the outcome of further evaluations. Of the nine vulnerabilities identified in Section 3.6.3 of the Submittal, five were addressed by specific planned improvements. The table on the following page indicates the planned improvements to the Kewaunee plant systems and/or operating procedures.

The improvements noted in the following table were identified by the IPE and are consistent with the vulnerability screening and importance evaluations discussed in Section 3.4 of the Submittal. As indicated by this table, the documentation indicates that the licensee is in the process of implementing these changes. Four areas of vulnerability other than those noted in the table were stated in the Submittal to be under review for possible future improvement. These included (1) fail-open vs fail-closed position for a valve allowing normal makeup from the condensate storage tank to the condenser, (2) actions needed to improve the reliability of the turbine driven auxiliary feedwater pump, (3) replacement of instrument air compressors, and (4) actions needed to reduce lifting of charging pump relief valves. As noted above, the quantitative impact on risk reduction of making these additional modifications was not evaluated by the licensee.

PLANT IMPROVEMENTS INITIATED BY THE IPE			
Improvement	**Vulnerability Addressed**	**Related Initiating Event**	**Schedule or Status***
Performed leak testing of an additional four valves serving as a boundary between the reactor coolant system and a low pressure system.	#1 (b), P. 396 of Submittal	Interfacing System LOCA	Completed
Modify the normal position of two motor operated valves located on the low pressure safety injection line from open to closed.	#1 (a), P. 396 of Submittal	Interfacing Systems LOCA	Deleted**
Modify emergency operating procedure ECA 1.2 to improve guidance to the operators in identifying and mitigating an interfacing systems LOCA.	#2, P. 396 of Submittal	Interfacing Systems LOCA	Completed
Modify the swing direction of three doors separating the turbine building basement with areas containing safeguards equipment in order to reduce the likelihood of a turbine building basement flood propagation into these other areas.	#3, P. 396 of Submittal	Internal Flooding	Completed
Improved the inspection method for rubber expansion joints to identify possible flooding problems before they occur.	#4, P. 397 of Submittal	Internal Flooding	Completed
Modify emergency operating procedures to provide instruction for switching the power supply to bus 5162 in the event of the loss of either safeguards bus 5 or 6 in order to have power available to two instrument air compressors.	#5, P. 397 of Submittal	Loss of Offsite Power, Station Blackout	Completed

*Status as of September, 1994
**Subsequent analysis indicated that designated improvement would introduce additional safety problems; therefore, this change will not be implemented.

Of the four identified vulnerabilities not discussed in the table, one was resolved by further analysis, two have been addressed by specific plant improvements already completed, and one is still under evaluation to best define any additional changes to the plant design or operation needed to address this vulnerability.

The Kewaunee IPE Submittal states that the analysis model used took credit for all of the improvements noted in the foregoing table. However, the analysis results did not present a before and after picture, i.e., no quantitative measure of the risk reduction achieved by implementing these changes was presented. Most of the planned changes have been made; in some cases, however, the licensee is evaluating the effectiveness of procedural/maintenance changes to assess whether or not additional improvements might by needed. Further evaluation of certain of the planned improvements indicated that the downside risks were likely to be more significant than the benefits; therefore, certain of the improvements will not be implemented.

Based on our review, we conclude that the licensee has taken specific actions in response to the identified vulnerabilities. More specifically, the licensee has identified both physical and procedure modifications that are expected to enhance plant safety.

II.1.5 Licensee's Evaluation of the Decay Heat Removal Function

II.1.5.1 Reliability of the DHR Function and Consistency With Other PSA Findings

The Submittal states that the only defined vulnerabilities are those described in Section 3.4.3 relative to the reliability of the auxiliary feedwater system. Vulnerabilities relating to DHR found in Section 3.4.3 are:

- The analysis determined that the auxiliary feedwater system contributed approximately 32% to the total core damage frequency. Approximately 21% is directly related to the reliability of the turbine driven auxiliary feedwater pump.
- A diversion path exists that diverts condensate from the condensate storage tanks to the main condenser, thereby reducing the quantity available to the auxiliary feedwater pumps for secondary cooling. If the operator fails to isolate this path, caused by a valve failing open on loss of instrument air or control power, then the success of auxiliary feedwater in providing secondary heat removal is adversely affected.
- There have been numerous cases in which a charging pump discharge relief valve opens and diverts charging pump flow back to the volume control tank, thereby affecting the ability of these pumps to provide reactor coolant system makeup as well as reactor coolant pump seal injection for seal cooling.
- Vulnerabilities associated with the instrument air system may affect DHR reliability. The analysis has shown that instrument air ranks about sixth in terms of system importance in contributing to core damage. Based on

this assessment, the licensee judged that this system needs improvement.
- The internal flooding analysis determined that a major flooding event could occur as a result of the failure of a circulating water expansion joint at the main condenser. Routine inspections to accurately assess the material condition of these expansion joints were not conducted.
- Vulnerabilities resulting in interfacing systems LOCAs also affect DHR. These LOCAs could compromise either the high or low pressure safety injection systems.

II.1.5.2 IPE Consideration of Diverse Means of Decay Heat Removal

The benefits of diverse means of DHR were not discussed, however, the IPE considered feed and bleed and recovery of main feedwater. In addition, the licensee provided discussions of the four possible DHR methods for cooling the reactor core, which include:

- Secondary cooling through the steam generators with main feedwater or auxiliary feedwater providing the steam generator makeup.
- Bleed and feed cooling using the high-head safety injection pumps and pressurizer PORVs.
- ECCS injection and recirculation as provided by the SI and RHR systems.
- Shutdown cooling mode of RHR operation after the RCS has been cooled down and depressurized to RHR conditions.

Each of the four means of removing decay heat were discussed, including a description of the required system's operation, needed support systems, and important operator actions. The major contributors to the unreliability of each system were also discussed. Thus, the licensee performed an evaluation of alternative means to accomplish DHR at Kewaunee.

II.1.5.3 Decay Heat Removal Unique Features

The Kewaunee IPE Submittal stated that the following are unique features contributing to increased DHR reliability:

- High head safety injection pumps can inject at 2200 psig, which is a higher pressure than typical Westinghouse plants.
- Containment sump recirculation can be aligned to the high head safety injection, low head safety injection and containment spray pumps from the control room.
- Three auxiliary feedwater pumps independent of cooling water systems; two motor-driven and one turbine-driven for diversity. The service water system serves as a backup suction supply to the three pumps.
- Four safety related service water pumps for a single unit site.

- The chemical volume and control system has three positive displacement charging pumps which are independent of cooling water systems. One of the pumps is driven by a variable speed DC motor for speed control and is not dependent on instrument air for attaining maximum pump output.
- Two independent methods for maintaining reactor coolant pump seal integrity, seal injection from the charging pumps and thermal barrier cooling via the component cooling water system.

II.1.5.4 Conclusion Regarding Kewaunee Evaluation Decay Heat Removal

The information submitted by the licensee demonstrated an understanding of diverse means for accomplishing DHR, of the front-line and support systems needed, and of the important operator actions needed.

III. OVERALL EVALUATION AND CONCLUSION

The Kewaunee IPE provides assistance in understanding the actions needed and options available for prevention and mitigation of severe accidents, and for identifying plant vulnerabilities.

The Kewaunee IPE uses the standard small event tree/large fault tree methodology to perform the Level 1 analyses. Functional event trees were developed to define the possible accident scenarios for each specific initiating event. Fault trees were created for each front-line system identified in the logic of the associated event tree(s) and fault trees were created for important support systems.

The IPE Submittal indicates that the analysis performed reflected the current plant design. The design cutoff date was stated to be December, 1992. Plant walkdowns and other methods were employed to help assure that the models used reflected the actual plant design. However, the documentation does not indicate that this was done for all systems or on a consistent basis. The models employed took credit for some physical modifications and procedural changes which have not yet been implemented. The IPE model was updated to better reflect current plant configuration and operating procedure.

The Kewaunee IPE models and analysis were subjected to independent reviews by both plant staff familiar with the plant systems and plant operation, and by an external team with expertise in PRA methodology (Level 1 and Level 2).

The internal flooding methodology used in the Kewaunee IPE was described. The IPE review indicated that the Kewaunee flooding analysis was generally complete, with the exception of the effects of sprays, which were assumed to be of importance only if equipment susceptible to spray damage was located within a radius of 10 feet from the spray source. Similarly, it is not clear from the Submittal if sprays were considered to be credible for insulated low-energy water piping, or if leaks in these systems were treated only as drip sources. The information provided by the licensee is not sufficient to assess the importance of this particular assumption or the impact on core damage frequency of using a less optimistic basis for spray-induced failures.

Regarding treatment of accident initiators, the licensee has described the process used to identify initiators (including internal flood) and has taken into account both generic and plant-specific information. Furthermore, the initiating events appear to be consistent with those considered in other PRAs.

Our review indicates that the licensee considered dependencies and asymmetries between and among systems in a manner comparable to that used in other PRAs/IPEs.

The Kewaunee IPE used the MGL method to model common-cause failures within systems. It appears that only generic data were used; consequently, there are no reports of root-cause analysis of plant-specific data. The licensee stated that a review of Kewaunee-specific data did not reveal any common cause failures. Therefore, generic CCF factors were used as a basis and were modified to Kewaunee-specific design features and conditions. The CCF factor values used are consistent with those used in other PRAs. The Kewaunee IPE approach for dealing with CCF is somewhat simplified and optimistic in that certain component groups such as batteries and breakers were not treated as being susceptible to common cause failures.

The review of the event trees included in the Kewaunee IPE Submittal indicated that they were logically arranged. In general, we found the event trees to reflect the success criteria and to model accident sequences. However, a comparison the importance of RCP seal LOCAs between the Kewaunee and Point Beach IPEs points out differences in the application of the Westinghouse seal LOCA models by licensees.

The Kewaunee IPE identified the most probable core damage sequences and identified the dominant contributors to each sequence. Comparisons with PRA results developed for other PWRs were not presented. However, our comparisons with PRA results from Point Beach and Surry indicate that the Kewaunee overall CDF estimates are generally consistent with results developed for these PWRs.

Independent checks that we performed of important front-end/back-end interface conditions and parameters indicated that Kewaunee interfaces were treated in their IPE evaluations.

The Kewaunee IPE quantitatively evaluated the impact of integrated system and component failures on plant safety. Sensitivity studies were performed on operator actions (including recovery actions), common cause, test and maintenance and failure rates for particular system components. The sensitivity to analysis cutoff frequency was also assessed. The information provided indicates that the licensee gained an appreciation of system importance. Thus, the Kewaunee IPE presented an assessment of the impacts of vital parameters on the estimated CDF. The analysis used point or central estimate values to quantify accident sequences.

The techniques used in the Kewaunee IPE are generally consistent with those used in other PSAs, and plant-specific data are used for nearly all components and systems, and for maintenance unavailabilities.

The licensees treatment of common cause failure data is roughly comparable to the treatment provided by other PRAs/IPEs. Sources of data were identified, and the use of CCF data in the IPE was explained.

The analysis supports the licensee's definition of vulnerability with respect to core damage. The analysis technique included consideration of potential failures down to the component level. The licensee's definition of core damage vulnerability has been used to identify plant vulnerabilities and potential safety enhancements. However, the licensee's analysis is such that the quantitative benefit in CDF reduction from the improvements cannot be estimated.

The information provided by the licensee discussed vulnerabilities regarding decay heat removal and diverse means of achieving DHR. Four means of removing decay heat from the reactor core were discussed, including descriptions of the required system's operation, needed support systems, and important operator actions. The major contributors to the unreliability of each system were also discussed. Thus, the information provided by WPSC indicated that the licensee performed a evaluation of alternative means to accomplish DHR at Kewaunee.

The Kewaunee IPE does provide assistance in understanding the actions needed and options available for prevention and mitigation of severe accidents, and for identifying plant vulnerabilities.

IV. IPE EVALUATION AND DATA SUMMARY SHEETS

Completed front-end data sheets are attached. They are provided in the format and organization specified in the draft 'Step 1' Review Guidance Document.

DATA SHEETS
Kewaunee Data
Note: Date Sheets not Updated to Reflect Licensee Responses to NRC Questions

2.4 Information Assembly

List of Plants of Similar Design

The following plants are Westinghouse 2-loop designs [NUREG/CR-5640]:

Ginna
Point Beach Units 1 & 2
Prairie Island Units 1 & 2

3.1.1 Initiating Events

- List IEs and enter frequency and contribution to core damage (probability and percent). Identify plant-unique IEs.

Initiating Event Category	Frequency (Per Yr)	Core Damage Frequency (CDF) (Per Yr)	Percent Contribution to Plant CDF
Large LOCA	5.0E-4	1.9E-6	3%
Medium LOCA	2.4E-3	8.1E-6	13%
Small LOCA	5.1E-3	1.4E-5	21%
Steam Generator Tube Rupture	6.4E-3	5.3E-6	8%
Reactor Vessel Failure	3.0E-7	3.0E-7	0.5%
Interfacing Systems LOCA	1.5E-6	1.4E-8	< 1%
Transients With Main Feedwater	3.0	2.7E-6	4%
Transients Without Main Feedwater	1.4E-01	4.6E-7	0.7%
Large Steam/Feedwater Line Break**	2.5E-3	1E-7	0 2%
Loss of Offsite Power	4.4E-2	4.5E-6	7%
Station Blackout**	4.4E-4	2.6E-5	40%
Anticipated Transient Without Scram or Main Feedwater **	3.8E-6	6.9E-8	0.1%
Loss of Service Water System	1.2E-4	4.2E-7	0.6%
Loss of Component Cooling Water System	1.6E-3	2.8E-8	< 1%
Loss of 125V Emergency Bus	2.4E-3	2.1E-7	0.3%
Loss of Instrument Air	1.1E-4	2.1E-6	3%
Flood in Turbine Building Basement*	8.9E-5	3.2E-10	< 1%

Initiating Event Category	Frequency (Per Yr)	Core Damage Frequency (CDF) (Per Yr)	Percent Contribution to Plant CDF
Flood in Turbine Building Basement*	1.1E-4	4.0E-10	<.1%
Flood in Diesel Generator Room A*	5E-4	1.8E-7	0.3%
Flood in Diesel Generator Room B*	5E-4	5.8E-8	0.1%
Flood in Relay Room*	1.5E-4	4.0E-10	< 1%
Flood in Control Rod Drive Equipment Room*	1.5E-4	3.1E-10	<.1%

* Plant-unique initiating event ** Special cases of more general initiators

3.1.2 Front-Line Event Tree Review

Basis for Success Criteria

The Success Criteria used in the Kewaunee IPE are based on the UFSAR analyses, analyses performed for the IPE, and engineering judgement.

Functional versus Systemic Event Trees:

The Kewaunee IPE utilizes functional event trees.

HVAC Assumptions:

HVAC requirements were evaluated in the IPE. All safety systems potentially dependent on the availability of HVAC were identified in the dependency matrix (Table 3.2-3) and in related tables (Tables 3.2-4 through 3.2-15). The discussions in the Submittal do not indicate that any evaluations were performed to determine if safety system equipment could function satisfactorily in the adverse conditions without the HVAC system. Also, although dependence on HVAC is indicated in Table 3.2-3, the nature of that dependence for any particular system is not discussed in any consistent manner from one system to another. Table 3.2-3 appears to be somewhat incomplete in that for at least one system (component cooling water) the dependence on HVAC is discussed in the system description but is not noted in Table 3.2-3. Finally, the portions of particular systems which fail on loss of HVAC (electric motors, motor control centers, electrical cabinets, etc.) are not identified. The Submittal does not indicate whether or not HVAC is needed for successful control room operation.

3.1.3 Special Event Tree Review

The Kewaunee IPE used special event trees to analyze steam generator tube rupture initiating events, and to analyze ATWS sequences. Also, a special event tree was developed to analyze large steam/feedwater line breaks.

RCP seal LOCA initiating events were categorized as small LOCAs and analyzed accordingly. Transient sequences in which an RCP seal LOCA developed were analyzed. RCP seal LOCAs were modeled to occur if both seal water injection and back leakage with CCW thermal barrier cooling were lost.

3.1.4 Support System Event Tree Review

Support system event trees were not used in the Kewaunee IPE. The support system fault trees were directly linked to the front-line fault trees.

The contractors supporting Wisconsin Public Service Company in the preparation of the Kewaunee IPE were Westinghouse Electric Corporation for the Level 1 PRA support and Fauske and Associates, Inc. for the Level 2 containment performance analysis.

3.2 System Analysis

3.2.2 Fault Trees

Fault trees developed for the Kewaunee IPE are listed in Section 3.2.1, "System Descriptions," of the Submittal. Fault trees were not provided in the Submittal. Typically multiple fault trees were developed for each system, depending on the type of function provided and/or the associated initiating event conditions. The systems for which fault trees are stated to be developed include:

Auxiliary Feedwater System
Component Cooling Water
Containment Air Cooling
Containment Isolation
Electrical Power (4160/480/120V AC, 120V DC)
High Pressure Safety Injection
Internal Containment spray
Low Pressure Safety Injection
Main Feedwater (including condensate)
Reactor Protection (including reactor trip system and ESF actuation system)
Service Water
Miscellaneous Systems, including:

- Chemical & Volume Control
- Reactor Coolant Charging
- Station and Instrument Air
- HVAC (Auxiliary building ventilation)

3.2.3 System Dependencies

Plant-Unique System Dependencies. No plant-unique dependencies were identified, and none were noted in the Submittal.

Important Plant Asymmetries. Asymmetries were not specifically discussed in the Submittal. Reviews of the UFSAR for Kewaunee and the system descriptions and drawings provided with the IPE Submittal indicated that there are relatively few asymmetries in the design or operation of most of the front-line or support systems. Where asymmetries occur, they appear to have been addressed in the evaluation. For example, auxiliary building basement cooling provides support for the safety injection and containment spray systems. There are four fan coil units (FCUs) which can be operated in any of five combinations for success, except for a particular pair operating together.

3.3.1 List of Generic Data

Source of Generic Data

NUREG/CR-4550, Rev. 1, Vol. 1
IEE-STD-500
NUREG/CR-2728
WASH-1400

3.3.2 Plant-Specific Data and Analysis

Source of Plant-Specific Data

Maintenance Work Requests
Incident Reports
Licensee Event Reports
Kewaunee Diesel Generator Reliability Study
Kewaunee Nuclear Plant Reliability Data System
Kewaunee Auxiliary Feedwater PRA

3.3.4 Common Cause Failure Analysis

Technique Used to Treat Common Cause Failures

Multiple Greek Letter (MGL) method

Level of Treatment

System (train) level or component groups

Most Significant Common Cause Failures

Event Identifier	Event Prob.	System or Component	Description in IPE Submittal
34---RHR-----CM	9.8E-4	RHR Recirculation	COMMON CAUSE FAILURE OF RHR
02----SW-----CM	6.2E-5	SW System	NO SERVICE WATER DUE TO COMMON CAUSE FAILURES
05B-SY1FAULT-CM	2.1E-4	AFW System	COMMON CAUSE FAILURES OF AFW SYSTEM
55--SY---SIAB-CM	1.5E-4	ESFAS Train	COMMON MODE FAILURE OF TRAIN A AND TRAIN B
35-OS1OS2EC4-CM	8.7E-5	RCS (Cooldown & Depressurization)	FAILURE DUE TO COMMON CAUSE FAILURES
02-SY-SWIE---CM	1.2E-4	SW System	SERVICE WATER SYSTEM COMMON CAUSE
34I--LPI-----CM	4.2E-4	LPSI Train	COMMON CAUSE FAILURE OF BOTH LPI TRAINS

Source of Common-Cause Failure Data

EPRI NP-3967, Classification and Analysis of Reactor Operating Experience Involving Dependent Failures
NUREG/CR-4780, Procedures for Treating Common Cause Failures in Safety and Reliability Studies, Procedural Framework and Examples

3.3.5 Quantification of Unavailability of Systems and Functions

Systems or Components with Noted Unusually High or Low Unavailability

None.

Source of Test and Maintenance Unavailabilities and Repair Rates

Kewaunee Technical Specifications
Operator logs
Engineering judgment

3.3.7 Quantification of Sequence Frequencies

Codes Employed in the Quantification Process

The codes used in the quantification process were stated to be as follows:

GRAFTER2 - Fault tree logic models
SIMON5 - fault tree initial quantification
WESCUT
WESLGE
WLINK - Fault tree linking and accident sequence quantification
COMPLNK - Importance analysis
WALT - Sensitivity analysis

Uncertainty Analysis or Sensitivity Analysis Performed?

Sensitivity and importance analyses were performed.

Sensitivity analysis scope: Sensitivity analyses evaluated were conducted on initiating event frequencies, operator actions, risk modeling, and plant design. Dominant initiating event frequencies were changed by reviewing system configurations, other data bases, possible alternatives that would prevent frequent occurrence of the initiating event, and the like, to determine a range of values for the frequency so that the variability of the core melt frequency could be assessed. Risk modeling sensitivity evaluations were stated to include changes to system success criteria, analysis assumptions, and other modeling criteria Design alternative studies included conceptual changes to systems whose failure contributes substantially to core melt frequency.

The Submittal states that sensitivity evaluations were performed for operator actions, common cause, test and maintenance and particular system components.

Sensitivity analysis method: Each sensitivity study was conducted by varying only one influence factor and holding all other factors constant. Failure rates were increased or reduced by a factor of 5 for high uncertainty aspects such as human reliability, common cause, and test and maintenance unavailability. Component failure rates were increased or decreased by a factor of two to test sensitivity. The components included in these evaluations were air operated valves, motor operated valves, and diesel generators. Fault tree cutoff probabilities were changed from 1.0E-12 to 1.0E-11 to assess sensitivity of the results to cutoff probability value selection.

The Kewaunee IPE Submittal discussed how each sensitivity evaluation was carried out and gave the results in terms of changes in core melt frequency.

The Kewaunee IPE Submittal also included an importance analysis of initiators, core melt sequences, cutsets, and components.

3.3.8 Internal Flooding

Methodology

Flood induced initiating events were identified. Critical components impacted by the flooding events were identified to determine if the postulated flood could initiate a reactor trip or endanger safe shutdown.

The flooding analysis included the following features:

- Flooding events that did not induce a reactor trip were screened from further analysis.
- Automatic system failures and human errors pertaining to the detection and isolation of the flood were considered.
- Findings were confirmed by a plant walkdown.
- Possible drain backflow was assessed by examining drain and trench drawings for the turbine and auxiliary building basements.
- The analysis was based on a 24 hour mission time.

Most significant assumptions include:

- No flood propagation was considered underneath doors with a gap of less than 1/8 inch.
- Doors opening away from a flood fail when water levels reach the 3 ft level.
- Doors opening toward a flood remain intact.
- Pipes leak and expansion joints fail catastrophically.
- Insulated pipes drip for a pinhole sized leak unless it is a high energy line in which case the insulation fails.
- Bare pipe leaks are spray sources with a 10 ft radius.
- Increased humidity effects on equipment were not considered.
- Only automatic reactor trips or immediate (within two hours of event) manual trips were considered.
- Reactor at power or in a hot shutdown mode at the time of event.
- Walls and barriers remain intact.

The screening process was based on whether or not the flood initiated a reactor trip and whether or not any critical components or systems were located within the flooded room. The screening assumed a maximum flood disabled everything in the flooded room. Only scenarios which initiated a reactor trip were considered for further analysis.

The appropriate event trees from the other internal initiating events were used to quantify the contribution of flooding to the core damage frequency by changing the initiating frequency to that of the flooding event and the failure probabilities of flood-effected components from their random failure values to the flood-induced failure value of one. The flooding-induced contribution to core damage was then computed for each sequence.

Contribution of Internal Flooding to Core Damage

After screening, six flooding scenarios remained for quantification. These are:

The IPE determined that no credible internal flood/spray scenario provides a significant contribution to the overall risk for Kewaunee. The largest contributor to core melt, caused by the failure of a service water expansion joint in DG room A, was evaluated even though its CDF is below the reportable limit.

Area of Flood	Source of Flood	Initiating Frequency (1/yr)	Core Damage Frequency (1/yr)
Turbine Building Basement	Condenser Circulating Water Expansion Joint (Winter Conditions)	8.9E-05	3.2E-10
Turbine Building Basement	Condenser Circulating Water Expansion Joint (Summer Conditions)	1.1E-04	4.0E-10
Diesel Generator Room A	Service Water Flex Connection on DG A	5.0E-04	1.8E-07
Diesel Generator Room B	Service Water Flex Connection on DG B	5.0E-04	5.8E-08
Relay Room	Potable Water Line	1.5E-04	4.0E-10
CRD Equipment Room	Service Water Line	1.5E-04	3.1E-10

Critical Internal Flood Areas

Flood areas are listed in the above table. DG room A was the most significant.

Most Critical Flood Sources

Source of greatest significance was the service water flex connection on DG A in DG room A.

3.4.1 Application of Screening Criteria

Form of truncation, probability frequency or cut set size

The Kewaunee IPE adopted the screening criteria of Appendix 2 of Generic Letter 88-20. The applicable criteria are as follows:

- Any systemic sequence that contributes 1E-7 or more per reactor year to core damage
- All systemic sequences within the upper 95 percent of the total core damage frequency.
- All systemic sequences within the upper 95 percent of the total containment failure frequency.
- Systemic sequences that contribute to a containment bypass frequency in excess of 1E-8 per reactor year.
- Any systemic sequences that the utility determines from previous PRAs or by utility engineering judgement to be important contributors to core damage frequency or poor containment performance.
- Sequences that, but for low human error rates in recovery actions, would have been above the applicable core damage screening criteria.

Definition of Core Damage

No specific definition of core damage was provided (e.g., core uncovery, fuel clad melt, etc.). Core damage was assumed to occur if short term cooling and/or long term cooling of the core were not achieved.

Total Core Damage Frequency

6.65E-5/yr.

Dominant Accident Sequences

(BNL data entry)

Dominant Contributors to Core Damage

The dominant contributors and their percent contribution are:

Station blackout	39.8%
Small LOCA	20.6%
Medium LOCA	12.3%

SG tube rupture	8.0%
Loss of offsite power	6.8%

Recovery Actions (Sequence Level)

Recovery actions considered in the Kewaunee IPE are discussed in Sections 3.4.2 and 3.4.4 of the Submittal. The recovery actions cited are listed below.

(a) **Cross-connections between buses/trains**
None

(b) **Restoration/repair of secondary side cooling**
Locally open manual valves on component cooling water heat exchangers.
Locally open feedwater bypass control valves and the corresponding steam generator PORV.

(c) **Alternate emergency power sources**
Provide power to charging pumps by aligning Technical Support Center (TSC) diesel generator.
Locally start TSC generator.

(d) **Other**
Locally open steam generator PORVs.
- Manually isolate RHR pumps.

3.4.2 Vulnerability Screening

Importance or Relative Ranking Provided?

No. Several vulnerabilities are discussed by are not rank ordered by importance.

Licensee's Definition of Vulnerability.

The definition given was "a feature in plant design, procedures, training, etc., which results in a contribution to core melt risk greater than what is expected."

Vulnerabilities (Identify Specifically DHR Related Vulnerabilities)

Valves in the low pressure safety injections lines connected to the reactor coolant system were normally in an open position, increasing the probability for an interfacing system LOCA. Also, motor operated and check valves in the RHR lines connecting to the reactor coolant system were not routinely leak tested.

Procedural guidance was not sufficient to assure that operators would properly diagnose interfacing system LOCA events.

Important safeguards equipment located in the turbine building basement was vulnerable to flooding from adjacent areas.

Instrument air systems were less reliable than desired due to lack of procedures to better assure that power is provided to air compressors from vital buses during station blackout and loss of offsite power events.

The auxiliary feedwater system (AFW) was prone to failure due to a diversion path that diverts condensate from the condensate storage tanks to the main condenser and reduce the quantity of condensate available to the AFW pumps for secondary cooling.

A review of system and component importance indicated that the AFW system, and particularly the turbine driven auxiliary feedwater pump, contribute substantially to core damage. Similarly, station and instrument air, particularly specific air compressors, are also substantial contributors.

The charging pump discharge relief valves have opened on numerous occasions to divert charging pump flow back to the volume control tank, affecting the ability of these pumps to provide reactor coolant system makeup and reactor coolant pump seal injection for seal cooling.

Plant Fixes in Response to Identified Vulnerabilities and Change in Core Damage Frequency if Known.

Improvements at Kewaunee initiated by the IPE are stated to be as follows:

- Perform leak testing of an additional four valves serving as a boundary between the reactor coolant system and a low pressure system.
- Modify the normal position of two motor operated valves located on the low pressure safety injection connecting to the reactor coolant system from open to closed.
- Modify emergency operating procedure ECA 1.2 to improve guidance to the operators in identifying and mitigating an interfacing systems LOCA.
- Modify the swing direction of three doors separating the turbine building basement with areas containing safeguards equipment in order to reduce the likelihood of a turbine building basement flood propagating into these other areas.
- Improved the inspection method for rubber expansion joints to identify possible flooding problems before they occur.
- Modify emergency operating procedures to provide instruction for switching the power supply to bus 5262 in the event of the loss of either safeguards bus 5 or 6 in order to have power available to two instrument air compressors.

Consideration of Plant life Extension in Proposed Plant Modifications (Y/N)

No. No considerations of plant life extension were evident in the Kewaunee IPE.

3.4.3 Decay Heat Removal

Methods of Removing Decay Heat

The following methods are available to accomplish DHR:

- High Pressure Injection
  - Accumulators Injection
  - Centrifugal Injection Pumps
- Low Pressure Injection
  - Residual Heat Removal System
- Feedwater
  - Main
  - Auxiliary
- Feed and Bleed Cooling
- Containment
  - Fan Coolers
  - Sprays
  - RHR heat exchangers provide recirculation heat removal

Ability of the Plant to Feed and Bleed.

Decay heat removal can be accomplished by feed and bleed if the main and auxiliary feedwater systems fail. Success requires the operator to recognize the need for action, start or verify operation of at least one of the high pressure safety injection pumps, and open at least one of the pressurizer PORVs.

Two PORVs are attached to the upper head of the pressurizer. Each has a design relief rate of 87,450 lbm/hr at 2335 psia (from NUREG/CR-5640).

Credit for Feed and Bleed?

Yes. Many of the accident sequences call for feed and bleed as a means of decay heat removal.

Credit for Recovery of Power Conversion System?

Recovering main feedwater was included in the analysis.

Steam Generator Dryout Time

The steam generator dryout time is sequence dependent, however, a lower limit of 30 minutes was specified.

Main Feedwater Trip on Reactor Trip (Y/N)

No. The main feedwater pumps continue to provide feedwater flow after reactor trip in the transients discussed in the Submittal.

Unique Front-end System Features

Important unique features include:

- High head safety injection pumps injecting at 2200 psig. Higher pressure than typical Westinghouse plants.
- Containment sump recirculation can be aligned to the high head safety injection, low head safety injection and containment spray pumps from the control room. RHR pumps are needed for sump recirculation with high head safety injection and/or containment sprays.
- Three auxiliary feedwater pumps independent of cooling water systems; two motor-driven and one turbine-driven for diversity. The service water system serves as a backup suction supply to the three pumps.
- Four safety related service water pumps for a single unit site.
- The chemical volume and control system has three positive displacement charging pumps which are independent of cooling water systems. One of the pumps is driven by a variable speed DC motor for speed control and is not dependent on instrument air for attaining maximum pump output.
- Two independent methods for maintaining reactor coolant pump seal integrity, seal injection from the charging pumps, and thermal barrier cooling via the component cooling water system.

6. PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

Important Insights Including Unique Safety Features

Important Insights

Important insights are discussed below under Implemented Plant Improvements and Improvements Under Consideration. The Submittal states that WPSC had the goal of developing a living PRA of Kewaunee that can be used as a tool in decision making for the life of the plant.

Unique Safety Features

See "Unique Front-End System Features" in Section 3.4.3 above.

Implemented Plant Improvements or Enhancements Stemming from IPE

Modify the swing direction of two doors separating the turbine building basement with areas containing safeguards equipment in order to reduce the likelihood of a turbine building basement flood propagating into these other areas.

Improved the inspection method for rubber expansion joints to identify possible flooding problems before they occur.

Planned Plant Improvements for which Credit has been Taken in the IPE

Perform leak testing of an additional four valves serving as a boundary between the reactor coolant system and a low pressure system (scheduled for implementation during 1993 refueling outage).

Modify the normal position of two motor operated valves located on the low pressure safety injection connecting to the reactor coolant system from open to closed (scheduled for implementation during 1994 refueling outage).

Modify emergency operating procedure ECA 1.2 to improve guidance to the operators in identifying and mitigating an interfacing systems LOCA (scheduled for implementation during summer of 1993).

Modify emergency operating procedures to provide instruction for switching the power supply to bus 5262 in the event of the loss of either safeguards bus 5 or 6 in order to have power available to two instrument air compressors (scheduled for implementation during summer of 1993).

Modify the swing direction of one additional door separating the turbine building basement with areas containing safeguards equipment in order to reduce the likelihood of a turbine building basement flood propagating into these other areas (scheduled for implementation during 1993 refueling outage).

Increase the minimum condensate storage tank (CST) volume to provide a 4 hour supply of condensate to the AFW system (rather than current 90 minute minimum supply) for coping with a station blackout event (scheduled for implementation during 1993 refueling outage).

Improvements Under Consideration

Assessments are being made concerning the fail safe position for a valve in the line providing normal makeup from the condensate storage tank to the condenser hotwell, and which can currently allow flow from the tank to the hotwell on loss of instrument air and/or loss of DC control power.

Evaluations are being performed of means to improve the reliability of the turbine driven auxiliary feedwater pump.

Two older, less reliable instrument air compressors are being replaced with two air cooled air compressors.

WPSC staff are investigating means to improve the reliable operation of the relief valves at the discharge of the charging pumps.

Calculations Performed in Support of Review

RHR Pump Temperature during Recirculation with One RHR HX In Service

The temperature of water from the sump to the RHR pumps during recirculation with CCW cooling of one RHR heat exchanger can be estimated. Table 6.2-7 of the UFSAR provides the following information:

Design UA: 0.68 E6 Btu/hr/F
Design Heat Duty: 26.0 E6 Btu/hr
Design Temperature Tube Side: 400 F
Design Temperature Shell Side: 350 F
Tube Side Design Flow and Temperatures:
1.0 E6 lb/hr, 160 F inlet,
133.5 F outlet
Shell Side Design Flow and Temperatures:
1.25 E6 lb/hr, 95 F inlet,
116.1 F outlet.

At steady state, assuming design conditions, the heat exchanger temperatures can be calculated as follows:

$$(1)\quad P = m_p c\,(T_{hi} - T_{ho})$$

$$(3)\quad P = UA\,\Delta T_{LM}$$

$$(2)\quad P = m_s c\,(T_{co} - T_{ci})$$

where P is heat input, m_p is primary flow rate, m_s is secondary flow rate, c is specific heat, and T's are temperatures of the primary (hot,h) and secondary (cold,c) inlet and outlet (I and o). T_{ci} is fixed at a desired value. UA is the overall heat transfer coefficient, and ΔT_{LM} is the log mean temperature difference defined as:

where $\Delta T_i = T_{ho} - T_{ci}$; $\Delta T_0 = T_{hi} - T_{co}$.

These three equations were solved using Mathematica 2.1. Sample output is provided in Figure A-1. At a heat load of 2.6 x 10^7 Btu/hr with T_{ci} of 95 F, the temperatures are

$$\frac{(\Delta T_i - \Delta T_o)}{Ln(\Delta T_i / \Delta T_o)}$$

as follows:

T_{hi} is 157 F

T_{ho} is 131 F
T_{co} is 116 F,
which agree with the design data presented earlier.

Figure 14.3-23 of the UFSAR implies that the switchover to recirculation occurs at about 4000 seconds. After 4000 seconds, decay heat is about 1.5% of full power, based on the Standard Review Plan decay heat curve, which is about 8.6 x 10^7 Btu/hr for Kewaunee, assuming 1650 MWt. Using design UA with T_{ci} of 95 F, the temperatures are as follows:

T_{hi} is 299 F
T_{ho} is 213 F
T_{co} is 163 F.

The sump temperature is 299 F, below the 400 F design rating for RHR, thus it is possible that RHR will not exceed allowable temperatures. This scoping calculation did not account for changes for UA with temperature and did not consider fouling of the RHR heat exchanger.

Figure A-2 provides T_{hi} for various times following shutdown with on RHR heat exchanger in operation, using decay heat from the Standard Review Plan, and T_{ci} of 95 F. This figure was produced with Stanford Graphics 2.1, using the output of the previously described Mathematica 2.1 calculations.

Kewaunee Seal LOCA Data Compared to Other IPE/PRAs

We compared data used for seal LOCAs in the Kewaunee IPE to data used in the Point Beach IPE and in the Surry NUREG/CR 4550 PRA. This comparison was performed with the aid of Mathematica.

The following data were used in the IPE for core uncovery due to a seal LOCA considering recovery of offsite power:

2.83E-2 by 2 hours with or without primary depressurization
7.62E-2 by 9 hours without primary depressurization
7.07E-2 by 11 hours with primary depressurization.

This data is for O rings not qualified for high temperature, and is based on the Westinghouse probabilistic seal LOCA model.

The seal LOCA data used in the Point Beach IPE was based on the Westinghouse seal LOCA model for unqualified elastomers. [WCAP 10541] The following information was available to us:

$$y(t) = 7.7\times10^{-4} e^{0.860\,t} \quad (pressurized)$$

$$y(t) = 3.3x10^{-4} e^{0.748\,t} \quad (depressurized)$$

Here y(t) is the probability that a seal LOCA occurs and leads to core uncovery by time t. Non-recovery of offsite power was modeled as:

The probability of core uncovery by time t due to a seal LOCA considering recovery of offsite power is given as:

$$G(t) = 0.61\, e^{-0.39\,t}$$

Table D-5-3 of the NUREG/CR 4550 PRA for Surry provides the data used in that PRA for modeling a seal LOCA; the data are based on expert solicitation.

$$P(t) = \int \frac{dy(t)}{dt} G(t)\, dt$$

Using Mathematica, we compared the data for seal LOCAs for the three IPE/PRAs as summarized below.

Figures A-3 and A-4 compare data for the seal LOCA models used in the Kewaunee IPE, Point Beach IPE, and the NUREG/CR-4550 PRA for Surry. Data for two cases are given: primary depressurized and primary pressurized. The data given are the probability that a seal LOCA occurs and leads to core uncovery as a function of time, considering recovery of offsite power that restores seal cooling, assuming that all seal cooling was lost to a tripped RCP at time zero. Both the Kewaunee and Point Beach IPE data for seal LOCA data used in the IPEs are based on the Westinghouse seal LOCA model for unqualified elastomers. It is notable that the data used in the Point Beach IPE predicts a lesser likelihood of core uncovery at times less than about 8 hours (pressurized) and 12 hours (depressurized) than the data used in the Kewaunee IPE, even though both data are based on the same model and both plants are similar.

Figure A-1. Sample Solutions for Heat Exchanger Temperatures

In[5]:=

```
Tci=95
P=2.6*10^7
UA=6.8*10^5

FindRoot[{P==10^6*(Thi-Tho),
          P==1.25*10^6*(Tco-Tci),
  P==UA*((Tho-Tci)-(Thi-Tco))/Log[(Tho-Tci)/(Thi-Tco)]},
  {Thi,200},{Tho,150},{Tco,120}]
```

Out[5]=

```
95
```

Out[6]=

```
      7
2.6 10
```

Out[7]=

```
680000.
```

Out[8]=

```
{Thi -> 156.694, Tho -> 130.694, Tco -> 115.8}
```

In[15]:=

```
P=8.6*10^7

FindRoot[{P==10^6*(Thi-Tho),
          P==1.25*10^6*(Tco-Tci),
  P==UA*((Tho-Tci)-(Thi-Tco))/Log[(Tho-Tci)/(Thi-Tco)]},
  {Thi,200},{Tho,150},{Tco,120}]
```

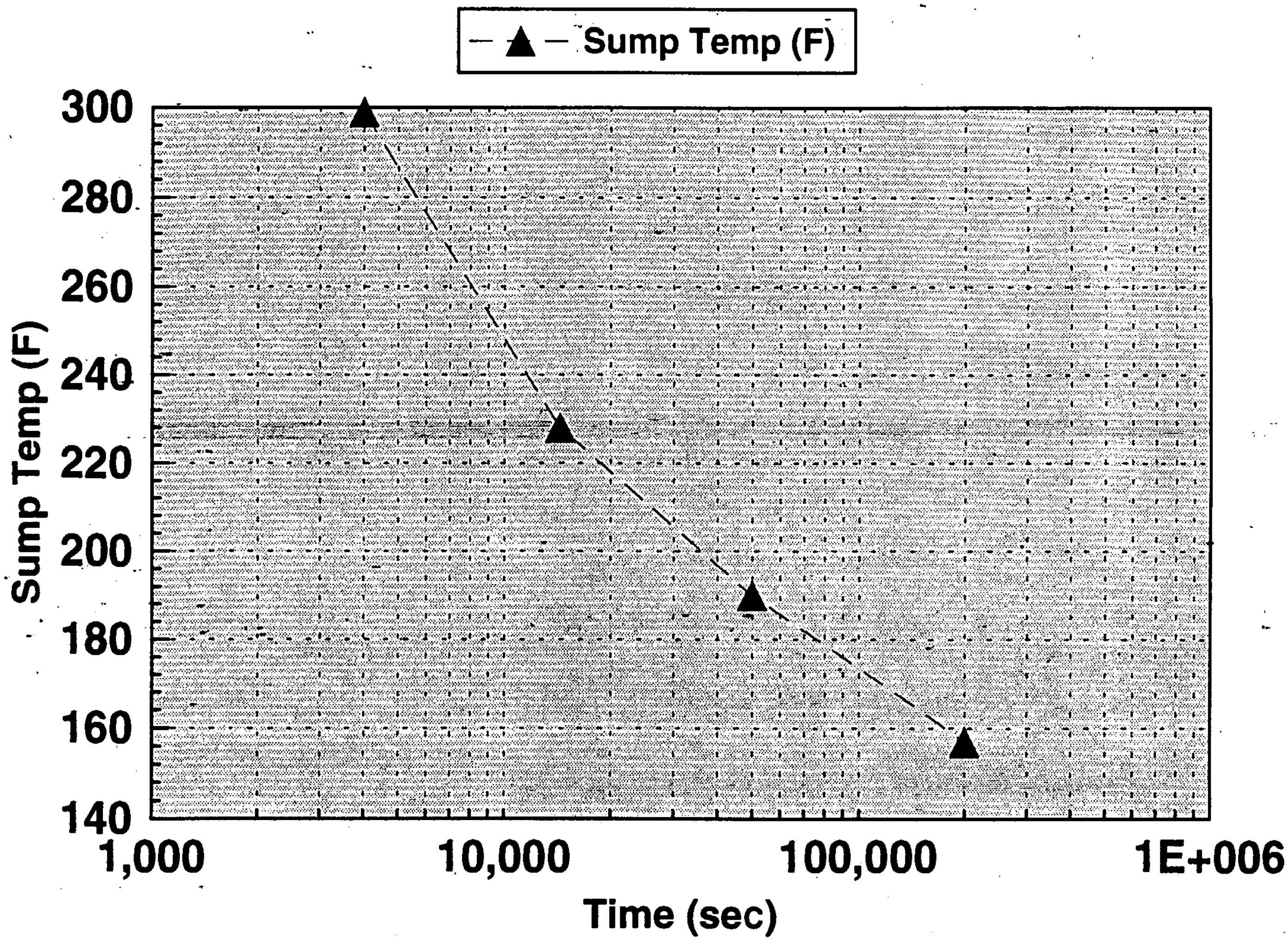
Out[15]=

```
      7
8.6 10
```

Out[16]=

```
{Thi -> 299.065, Tho -> 213.065, Tco -> 163.8}
```

Figure A-2. T_{hi} Values at Various Times



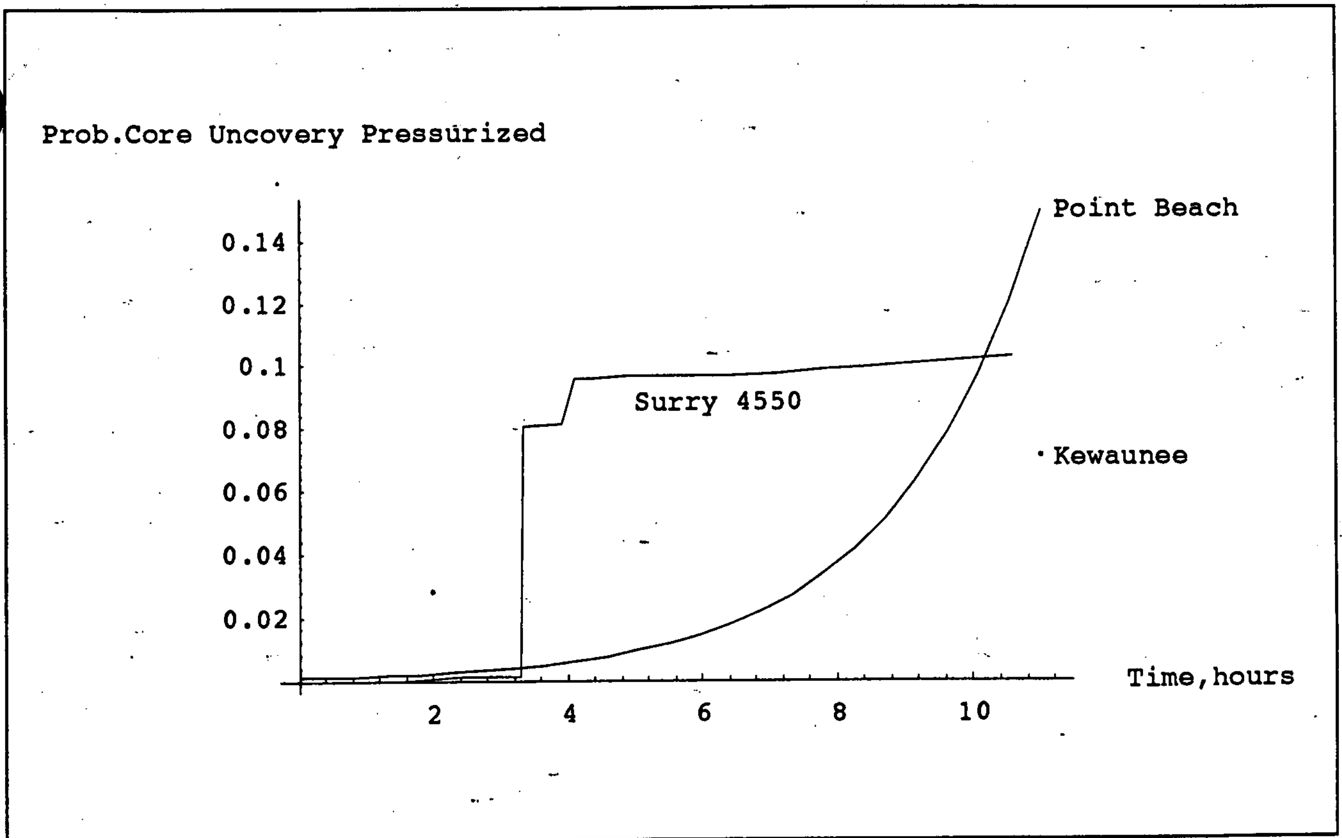


Figure A-3. Seal LOCA Models (pressurized)

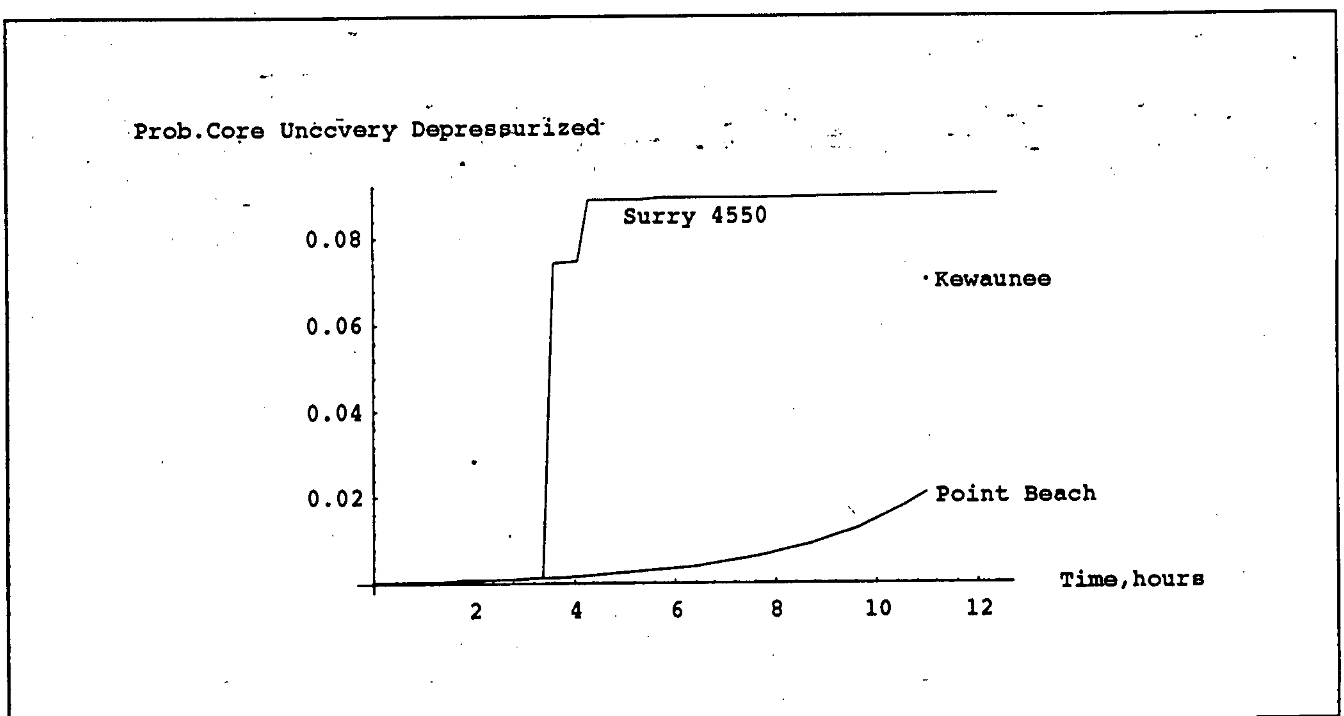


Figure A-4. Seal LOCA Models (depressurized)

APPENDIX B

KEWAUNEE NUCLEAR POWER PLANT INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)