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Watts Bar Unit 1
Technical Evaluation Report on the IPE Front-End Submittal

Contractor Step 1 Audit Report
NRC-04-91-066, Task 12

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Prepared for the
Nuclear Regulatory Commission

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IPE-FE REVIEW
WATTS BAR UNIT 1

I. INTRODUCTION

This introduction chapter presents the process used by Science & Engineering Associates, Inc. (SEA) to review the front-end portion of the Tennessee Valley Authorities (TVA) Individual Plant Examination (IPE) submittal for the Watts Bar Unit 1 Nuclear Station. 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 analysis and back-end aspects of the Watts Bar IPE were performed by the NRC with contractual assistance from Concord Associates, Inc. and Scientech, Inc., respectively.

I.1 SEA Review Process

The evaluations presented herein are the result of a Step 1 review. We reviewed the licensee's process as documented in the Submittal, with respect to meeting the requirements of Generic Letter 89-20. Issues were identified and provided to the NRC for consideration for further discussion with the licensee. The NRC requested further information from the licensee on selected issues, and the licensee responded to these issues.

Since this review was performed, the licensee provided a revised IPE to the NRC. We briefly reviewed the revised Submittal, but that review is not reflected in the rest of this report which documents the review of the original Submittal. Our comments on the revised Submittal are as follows. The frequency for total loss of CCS as an initiating event was reduced by about two orders of magnitude based on more detailed analyses of the number of CCS pumps required to support normal operation at power; similarly, the frequency for the initiating event loss of train A of the CCS decreased by about an order of magnitude. The common cause failure factors for ERCW pumps were updated to reflect industry experience over the last five years; this lowered the frequency of initiating events associated with loss of ERCW by about an order of magnitude. The frequency

for loss of offsite power was originally based on generic data, and was revised by using plant specific data to bayesian update the generic data; this lowered the frequency of loss of offsite power by about a factor of 2. Based on MAAP analyses, the success criteria for feed and bleed were changed from 2 PORVs to 1 PORV. The probability of operator action to trip RCPs after loss of cooling was increased slightly to reflect changes in procedures. The revised IPE still assumes that failure to trip a running RCP after loss of cooling results in a seal LOCA in 10 minutes. These changes lower the overall CDF from 3.3E-4 to 8.0E-5/ Ry yr.

I.1.1 Review of UFSAR and Tech Specs

The NRC provided the Watts Bar IPE submittal to SEA in January 1993. SEA began work on the Watts Bar early February 1993.

In February 1993, selected portions of the latest (updated) Safety Analysis Report (USAR) for Watts Bar were copied and made available to SEA's lead analyst on this Step 1 audit. These copies were made from up-to-date documentation provided by the NRR Project Manager. These documents were reviewed as needed during the course of the overall audit. No final technical specifications have been approved at the time of review.

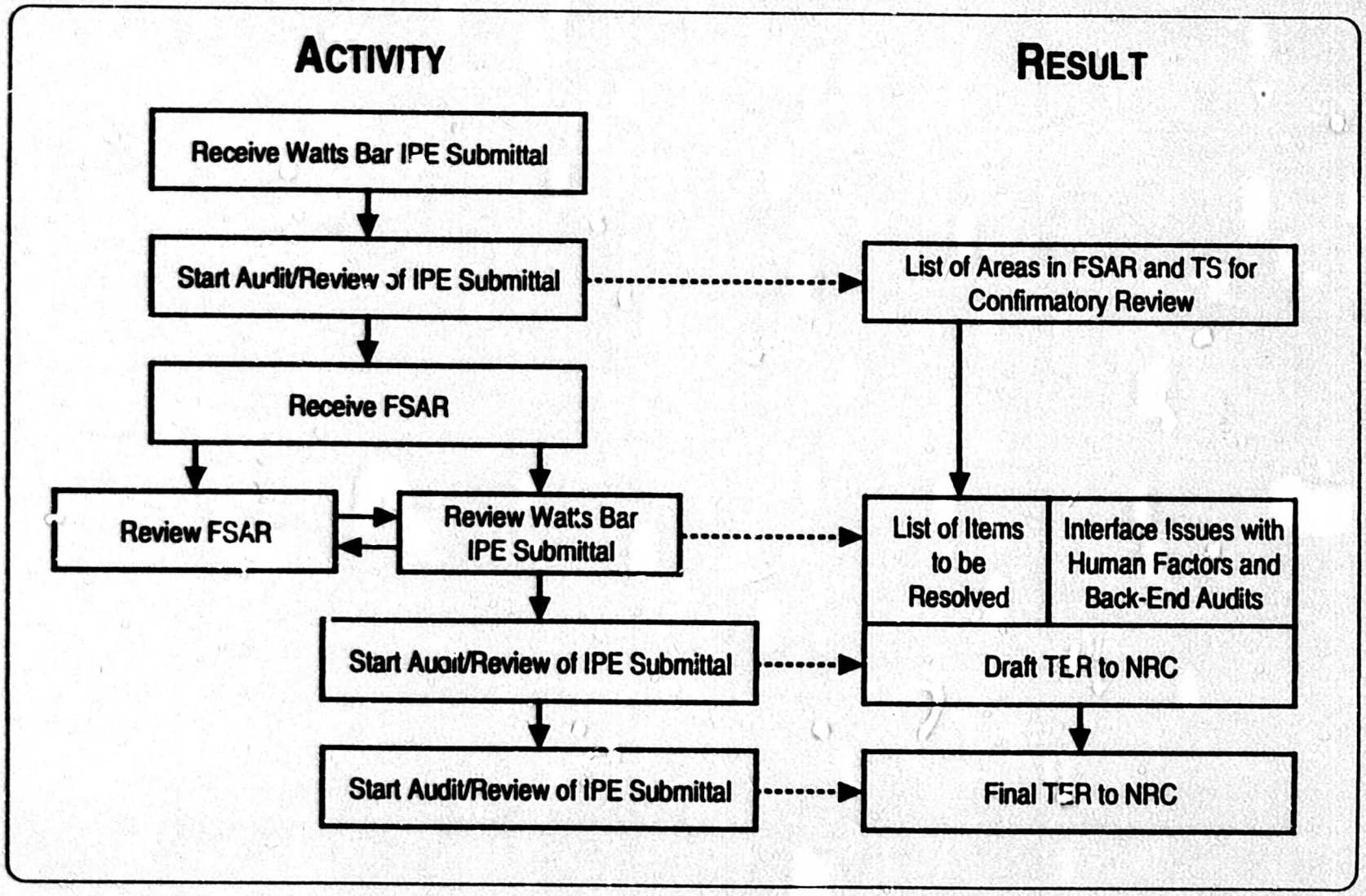

Figure I.1-1. SEA Step 1 Audit/Review for Watts Bar IPE-FE

I.1.2 Review of IPE Submittal

A detailed review of the IPE submittal for Watts Bar was accomplished. The effort incorporated a complete review of all aspects of the front-end issues called for in the statement of work (SOW) for this task and NUREG-1335. In addition, the guidance provided in the "Draft Step 2 Review Guidance Document," dated May 19, 1992, was utilized.

I.2 Watts Bar IPE Methodology

The Watts Bar IPE uses the large event tree/small fault tree methodology to perform the front end analyses. A support system event tree was used to evaluate support systems and their impacts on front-line systems. Recovery actions are considered and common mode failures are incorporated into the fault tree models. Importance analyses were performed on selected parameters.

I.3 Watts Bar Plant

The Watts Bar Plant is a two-unit site located in Rhea County, Tennessee, approximately 50 miles northeast of Chattanooga. Each unit has an initial power rating of 3411 MWth. Unit 1, which is currently under construction, is the lead unit for the plant. Unit 1 is a Westinghouse four-loop, pressurized water reactor with an ice condenser containment.

I.3.1 Similar Plants and PSAs

The following plants are also Westinghouse four-loop PWRs with ice condenser containments

- Sequoyah
- Catawba
- McGuire

Probabilistic Safety Studies (PSA) for plants similar to Watts Bar include the Sequoyah NUREG-1150 study.

I.3.2 Unique Features

The Watts Bar IPE submittal identified several beneficial plant and procedural features. The features that are considered unique are discussed below:

1. The operators are directed to begin refill of the CST as soon as it drops below the technical specification limit (Procedure E1, Loss of Reactor or Secondary Coolants).
2. The CST has a relatively large capacity (370,000 gallons).
3. A standby electric-driven main feedwater (MFW) pump is available.
4. The power supply to the shutdown buses is not transferred after a unit trip.
5. Both steam generator level indication and AFW turbine pump control are supplied DC power from the opposite unit.
6. The CSS and RHR systems utilize different sump isolation valves.
7. The fifth station diesel generator has no procedures governing its usage.
8. Safety Grade Air System.

II. SEA REVIEW FINDINGS

This section discusses the Watts Bar IPE review findings. The organization of the findings strictly follows that specified under Subtask 1 of the statement of work. Task topic headings are in bold type.

II.1 Review and Identification of IPE Insights

II.1.1 General Overview of Front-End Analysis

II.1.1.1 Completeness of Submittal

The Watts Bar IPE submittal was, with minor exceptions, organized and presented strictly according to Table 2-1, "Standard Table of Contents for Utility Submittal," provided in NUREG-1335. The submittal contains the type and level of detail requested in NUREG-1335.

II.1.1.2 Description and Justification for Methodology Used

The Watts Bar documentation indicated that the methodology was consistent with NUREG/CR-2300. The IPE used the large event tree/small fault tree method. Support system event trees were used. The methodology used is briefly summarized in Section 2.3 of the submittal, and is more thoroughly discussed in appropriate locations in the documentation.

Initiating events were identified by reviewing both other PRAs for similar plants and PWR generic experience. All major plant systems were subjected to a FMEA to identify important support system failures.

Event sequence diagrams were used to help document required system and operator response to initiating events and to assist in the plant modeling through detailed event trees. System interdependencies were modeled in the event trees, as were human

interactions. Systemic event trees were used where the tree top events represent system responses and operator actions.

The analysis used a front-end/back-end interface boundary that provides for an integrated treatment of the systems in the Level 1 event trees and reserves the level 2 analysis primarily for the treatment of phenomenological issues associated with severe accidents and post-core damage accident management.

System fault trees were not included in the submittal. However, the fault trees appear to have been developed down to the component level. The dependencies among systems were documented in a system dependency matrix and appear to have been considered in the modeling. The analysis used an approach that consistently treats common cause events in accordance with NUREG/CR-4780 and includes consideration of test and maintenance alignments. The fault tree logic models are stated to have been quantified using the RISKMAN Software.

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

Section 1.4.1 of the submittal states that the Watts Bar IPE was based on the plant configuration as it existed on December 1, 1991. Two plant changes completed after the freeze date were included in the model. These changes are the shutdown boards continuous feed from an offsite source and improved plant procedures relating to the verification of ERCW flow to the diesel generators.

Section 2.3.3 of the submittal discusses the initial plant walkdowns performed by the PRA team to visually inspect key areas of the plant and major plant systems. The PRA team also held meetings with the cognizant engineers and plant operations personnel to confirm the analysts' understanding of the plant and to obtain agreement on the dependency matrices.

The preparation of system notebooks utilized the Final Safety Analysis Report (FSAR), plant procedures, plant drawings, and "other" qualitative information.

II.1.1.4 Internal Flooding Methodology

The Watts Bar flooding analysis methodology included the following steps:

1. Plant Familiarization - Key plant design information was reviewed.
2. Flood Experience Review - Flood data collected from Nuclear Power Experience (S.M. Stoller Corporation, Nuclear Power Experience, updated monthly) was reviewed to ensure familiarity with actual flood events, their locations within the plant, and their causes.
3. Evaluation of Flood Sources - Using the plant design information and a general knowledge of plant layout, major flood sources and their locations were identified.
4. Evaluation of Plant Locations - Using plant design information such as arrangement drawings, internal flood studies, and information from the evaluation of flood sources, the buildings most important to risk were identified.
5. Plant Walk Through - A walk through was performed to collect additional information and to confirm previous documentation and judgments on flood sources, their potential impact, propagation paths, and detection.
6. Scenario Quantification - Based on the above steps, scenarios were postulated, evaluated and quantified for use as initiating events. The effect on plant systems from the initiator is also evaluated.
7. Risk Model - To develop flooding core damage sequences, the flood scenarios were included as initiating events to the transient event tree.

It should be noted that flood induced failures stemmed from submergence. Spray induced effects "were judged to be localized", and bounded by the screening assumption.

II.1.1.5 Utility Peer-Review

Section 5 of the Watts Bar IPE submittal describes the project review process followed by TVA. The Risk Assessment Staff (RAS) coordinated the review of the IPE. PRA training sessions were held for site personnel by RAS staff. Site personnel were utilized in the independent review process. The personnel performing the IPE presented information to the site personnel. Comments from site personnel were incorporated into the IPE as necessary.

In coordination with the Watts Bar personnel, Dr. Ian B. Wall conducted an independent peer review of the IPE.

II.1.2 Review of Accident Sequence Delineation and System Analysis

II.1.2.1 Identification of Initiating Events and Related Appendices

The initiating event categories for the Watts Bar IPE were identified using several approaches; a comparison with category events from previous PRAs and other industry studies, a failure modes an effects analysis (FMEA) of the plant systems, a review of the Final Safety Analysis Report (FSAR), discussions with plant operators about specific postulated events, and a review of plant trips that have occurred at TVA's Sequoyah Nuclear Plant.

The IPE submittal states that the identification of initiating events was in part based on a review of initiating event lists for other Westinghouse plants. These lists were obtained from the following documents.

1. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment," prepared for Pacific OAS and Electric Company, Vols 1-9, PLG-0637, July 1988.
2. PLG, Inc., "South Texas Project Probabilistic Safety Assessment, Summary Report," Prepared for Houston Lighting & Power Company, PLG-0700, Vols. 1 and 2, April 1989.

3. Bertucio, R. C., et al., "Analysis of Core Damage Frequency: Sequoyah, Unit 1 Internal Events," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-4550, Vol. 5, Rev. 1, April 1990.

4. EG&G Idaho, Inc. "Development of Transient Initiating Event Frequencies for use in Probabilistic Risk Assessments," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-3862, May 1985.

5. U.S. Nuclear Regulatory Commission, "Reactor Safety Study, An Assessment of Accident Risk in U.S. Nuclear Power Plants," WASH-1400, NUREG-75/014, 1975.

6. American Nuclear Society and Institute of Electrical and Electronic Engineers, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute, NUREG/CR-2300, April 1983.

7. PLG, Inc., Westinghouse Electric Corporation, and Fauske & Associates, Inc., "Indian Point Probabilistic Safety Study," prepared for the power authority of the state of New York and Consolidated Edison Company of New York, Inc., March 1982.

8. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," PLG-0500, Vol. 6, PWR Initiators, Proprietary, August 1989.

The Watts Bar IPE grouped initiating events, by similarity of plant response, into initiating event categories. The list of categories developed for the Watts Bar IPE is presented in Table 3.1-1 of the submittal. No discussion of the initiating events grouped into categories was provided. Listed below are the transient initiator categories for Watts Bar along with those for Sequoyah (NUREG/CR-4550) and for the McGuire IPE.

TRANSIENT INITIATION CATEGORIES		
Watts Bar IPE	Sequoyah	McGuire

Reactor Trips	Loss of Offsite Power (T1)	Plant Trip
Core Power Excursion	Loss of Power Conversion System (T2)	Loss of Load
Turbine Trip	Turbine Trip w/PCS Initially Available (T3)	Loss of Off-Site Power
Inadvertent Safety Injection		Loss of Main Feedwater
Total Loss of Main Feedwater		Excessive Main Feedwater
Partial Loss of Main Feedwater		Secondary Line Break
Loss of Condenser Vacuum		Inside Containment
Excessive Feedwater		Feedwater Line Break
Inadvertant Closure of One MSIV		Outside Containment
Inadvertant Closure of all MSIVs		Steam Line Break
Loss of Primary Flow		
Steam Line Break Outside Containment		
Steam Line Break Inside Containment		
Inadvertent Openings of Main Steam Relief Valves		

Listed below are the loss of coolant initiator categories for Watts Bar along with those for Sequoyah (NUREG-CR-4550) and for the McGuire IPE.

LOCA Initiator Categories		
Watts Bar IPE	**Sequoyah**	**McGuire**
Excessive LOCA (Vessel Failure)	Small LOCA (<2")	Small LOCA (3/8" - 2")
Large LOCA (>6")	Medium LOCA (2" - 6")	Medium LOCA (2" - 5")
Medium LOCA (>2" - 6")	Large LOCA (<6")	Large LOCA (75")
Small LOCA (nonisolable)	ISLOCA	SGTR
Small LOCA (available)		Vessel Rupture
SGTR		ISLOCA
ISLOCA-RHR Injection Path		
ISLOCA-RHR Suction Path		

Table 3.1.1-1 of the submittal provides a list of plant-specific support system initiators that were generated by considering the plant impact of failures of major systems. The underlying evaluation is a system level failure modes and effects analysis. The FMEA performed appears complete in both the selection of systems analyzed and the effect of the support system initiator on other plant equipment.

The front-line system success criteria for the Watts Bar IPE were presented in Table 3.1.1-3 of the submittal. The IPE stated that the success criteria for the major plant safety functions were determined through engineering knowledge of the plant and a careful review of IPE references 3.1.1-3 through 3.1.1-10. The Watts Bar criteria were checked against those from the Sequoyah-NUREG/CR-4550 analysis. The criteria match except for the late core heat removal requirement for medium LOCAs: Sequoyah requires 1/4 HPR and 1/2 LPR for success, Watts Bar requires only 1/2 LPR.

As described in Section 3.3.8 of the IPE submittal, the process used to identify internal flood initiators and potential dependencies consists of:

- Review of details of the plant layout to familiarize analysts with the location of potential flood sources and pathways available for propagation,
- Review of causes and effects of actual flooding at nuclear power facilities from Nuclear Power Experience, and
- Plant Walkdowns.

The IPE calculates a low frequency for core damage due to interfacing LOCAs; the frequency of core damage with containment bypass is about 4×10^{-8} per year. Appendix E of the IPE indicates that this low frequency is attributable to consideration of operator action, and realistic consideration of relief valve capacities and true failure pressures of components exposed to pressures beyond design. The latter effect is stated to be most important. The best estimate analysis for failures of low pressure components exposed to higher-than- design pressures requires further confirmatory evaluation.

II.1.2.2 Review of Front-Line and Support Systems Analysis

The systems modeled or considered in the Watts Bar IPE are listed below.

- Electrical Power System
- Essential Raw Cooling Water System
- Component Cooling System
- Plant Compressed Air System
- Chemical and Volume Control System
- Condensate and Feedwater System
- Engineered Safety Features Actuation System
- Reactor Protection System
- Auxiliary Feedwater System
- Main Steam System
- Residual Heat Removal System
- Safety Injection System
- RCP Seal Injection and Thermal Barrier Cooling
- Pressurizer Power-Operated Relief Valves and Safety Valves
- Steam Generator Isolation
- Containment Spray System
- Containment Systems

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

Section 3.2.3 of the Watts Bar IPE submittal specifically addresses dependencies. This type of information is also presented in the individual system descriptions. Table 3.2-3 shows how failure of support systems affects equipment in other support systems. Table 3.2-4 shows how a failure of support system equipment affects front-line system trains of equipment.

II.1.2.4 Treatment of Common Cause Failures

The contribution to system unavailability from common cause dependent failures is treated by the Multiple Greek Letter (MGL) method according to the general methodology described in NUREG/CR-4780.

The IPE modeled those common cause failure modes within each system that satisfied the following screening criteria:

- When identical, nondiverse, and active components are used to provide redundancy, they should be considered for assignment to common cause groups, one group for each identical redundant component.
- The likelihood of common cause events linking diverse components in the system can be assumed to be negligible compared to identical, nondiverse, and active components that are present in the system.
- When diverse major components serving the same function have parts that are identically redundant, the components should not be assumed to be fully independent.
- When each redundant leg of a system contains one or more active components, the contributions due to both independent and common cause events involving passive components are generally insignificant in the calculation of system unavailability.
- In redundant systems in which no identical active components or parts can be identified, no common cause grouping need be attempted.

II.1.2.5 Review of Event Trees

II.1.2.5.1 Transient and Small LOCA Event Tree

The GENTRANS event tree models transients, small LOCAs, and ATWS scenarios.

Feedwater line breaks were not modeled in this tree. It should be noted that the effect of a feedwater line break is different than that from a steam line break. Feedwater line breaks lead to undercooling due to loss of water, while steam line breaks lead to overcooling due to depressurization. Also, the hardware for isolation of feedwater line breaks, such as check valves, is different from the hardware for isolation of steam line breaks, MSIVs.

Instrument tube LOCAs were not explicitly considered. These LOCAs are unique due to their location such that uncovery of the break with subsequent steaming out the break is not possible without the core being totally uncovered. In many plants, the size of a break in an instrument tube is within the capability of the normal makeup system and, thus, is not a LOCA; however this aspect was not noted in the IPE for Watts Bar.

The success criteria for AFW for ATWS is 2 MD pumps each feeding 2 SGs, or the TD pump feeding 4 SGs; the basis for this success criteria was not indicated in the IPE. The basis for the feed and bleed success criteria was not indicated in the IPE.

It should be noted that the IPE model does not address Pressurized Thermal Shock (PTS), and no quantitative evaluation supporting screening out PTS concerns is provided, although PTS was qualitatively screened out. [IPE page 3.1.2-23]

The ATWS model does discuss early in life conditions when the coolant is highly borated, and the moderator temperature coefficient may not be sufficiently negative to enable an ATWS to be mitigated.

The IPE did not model boration following a main steam line break. Without boration, the primary may return to excessive power as the primary cools below hot zero power. Page 3.1.2-32 of the IPE acknowledges this limitation in the model and claims the impact is not important. Also, the isolation of feedwater to the broken generator was not modeled, and this can also cause problems with power level. The IPE also did not address potential

degradation of systems exposed to harsh environment following a main steam line break outside containment; however, these considerations should have been considered in the licensing analyses.

The IPE success criteria for a small LOCA (less than 2 inches) is one of four CVCS/SI pumps, and for a medium LOCA (2 to 6 inches) the success criteria is two of four pumps. The FSAR success criteria is one CVCS pump and one SI pump. The IPE success criteria for small and medium break LOCAs deviates from the FSAR success criteria.

For example, consider a two inch break. The IPE success criteria indicate that no steam generator cooling is needed for this size break (the minimum medium size LOCA). Generally credit is taken for cooling using the initial stored inventory in the steam generators, but is not taken for continued feedwater to the steam generators.

$$\dot{Q} = \dot{m}(h_o - h_i) \qquad (Eqn.A)$$

for a mass and energy balance with no heat removal by the steam generators and steaming out the break, where Q is the decay heat, m is mass flow of water in, and steam out, h_o is specific enthalpy of steam out, and h_i is specific enthalpy of water in. Figure II.1.2.5-1 shows steam mass flow rates for a 2-inch hole and pump head flow rates for the cases of 1 CVCS pump, 1 SI pump, 1 CVCS and 1 SI pump, and 2 SI pumps, using data from Section 6.3 of the FSAR. Using the match point for 1 CVCS and 1 SI pump, equation (A) implies that the decay heat is 0.022 of full power; using the match point for 1 SI pump the decay heat is 0.017 of full power; using the match point for 2 SI pumps the decay heat is 0.019 of full power. Based on the Standard Review Plan decay heat curve, 0.022 is reached at 25 minutes, 0.017 is reached at 50 minutes, and 0.019 is reached at 33 minutes. Assuming a steam generator dryout time of about 30 minutes, this implies that both a CVCS and a SI pump are sufficient for a 2-inch equivalent small LOCA; however, a single SI pump may be pushing the limit. (It should be noted that some Westinghouse four loop plants, such as South Texas, have a significantly higher

capacity SI pump with a higher shutoff head and, for these plants, the FSAR licensing analyses does not require the CVCS pump.)

It should be noted that the IPE does not model tripping of the RCP pumps when subcooling margin is lost during a small/medium LOCA. If the pumps are not tripped, the degree of core uncovery can be greater.

No credit is taken for RCS cooldown followed by accumulator injection and low head injection since no analyses of this strategy has been performed. However, the procedures direct the operators to perform this action if high head ECCS is not available.

The IPE claims that 300,000 gal. in the CST can maintain hot standby for 24 hours. Our independent calculations confirm that this is an appropriate assumption.

II.1.2.5.2 Other Event Trees

The IPE assumes that failure to switch from cold to hot leg injection following a large LOCA results in late core damage due to core flow blockage. [IPE page 3.1.2-12] Other PRA's have assumed that failure to switchover is not a problem, although they generally do not provide justification. The concern is that for a large LOCA at long times, the injected water can flow out the break and subcooled injected water only makes up what boils off the RCS. Boron precipitates from the water that is boiled off.

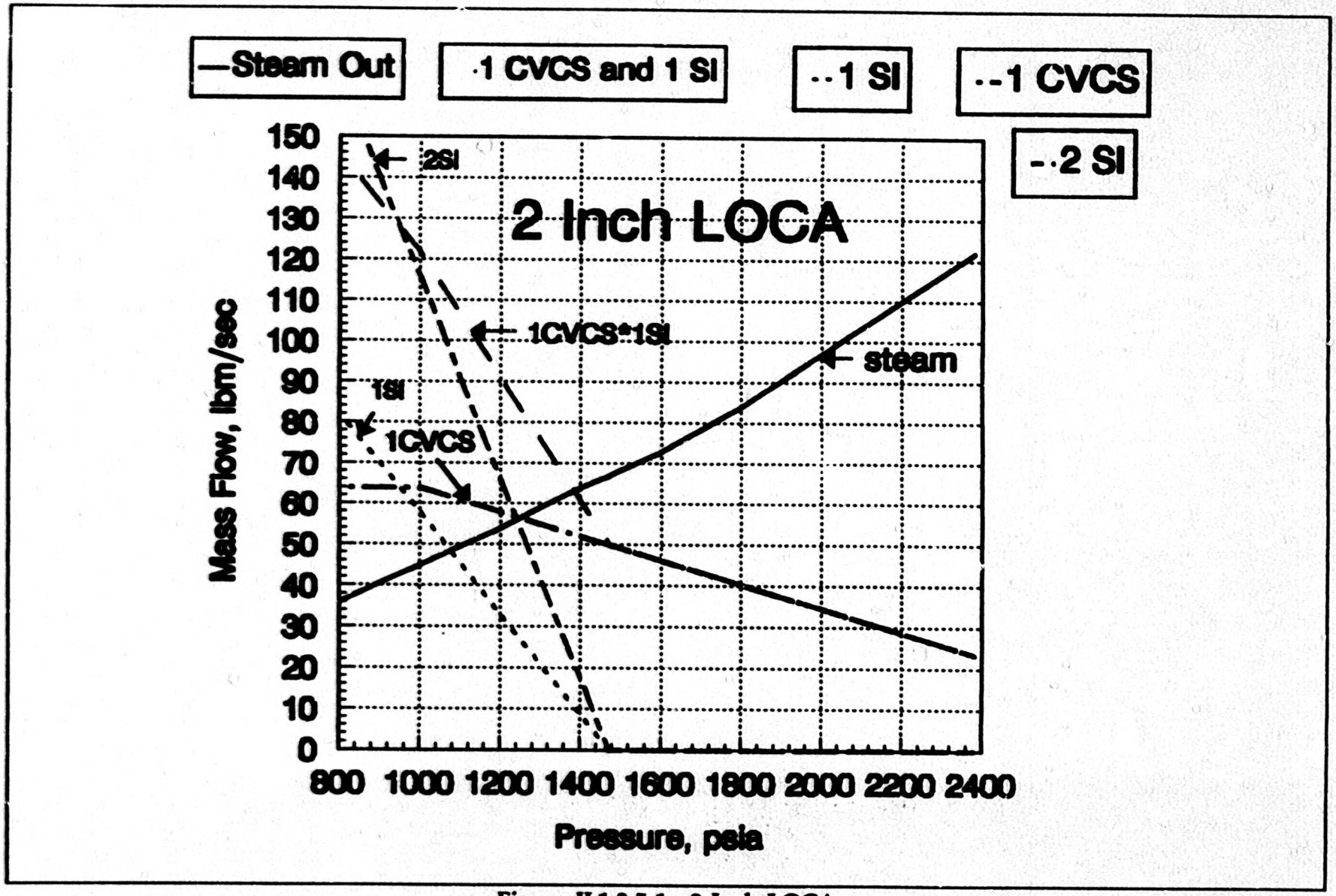

Figure II.1.2.5-1. 2 Inch LOCA

In the RECIRC event tree, no credit is taken for depressurizing the RCS to allow recirc from the sump without using high head pumps. [IPE page 3.1.2-72]

The electric power model does not include use of Unit 2 DGs for powering Unit 1 pumps, but does assume that Unit 2 diesels are always available for powering instrumentation and control at Unit 1. No discussion of the ability to cross tie Unit 2 DGs to Unit 1 is provided. The IPE does not model use of the fifth diesel generator, since it is not planned for unit 1 completion. The IPE does not discuss the ability to use 500 KV offsite power if both sources of 161 KV offsite power are lost. It appears that the 500 KV supply could be used if the main generator is disconnected (to prevent motoring the main generator).

The ESDs discuss the operator response to loss of RHR sump recirculation. [IPE, page 3.1.2-17] Two options are discussed: (1) refill RWST from other sources of borated water, such as the spent fuel storage pool, and (2) provide makeup using the containment spray to recirculate water back to the RWST from the containment sump. Other sources of borated water to refill the RWST were not considered. Option (1) was not credited due to time constraints. No success criteria for Option (2) was provided. No discussion of the radiological concerns associated with recycling containment sump water back to the RWST were addressed.

It should be noted that the event trees do not clearly discuss the impact of loss of containment cooling on the ability to recirc from the sump to cool the core. This could impact the relative timing of core damage and containment failure for accidents in which recirculation from the sump is available, but containment cooling is not available. (This is the same issue that we have discussed in many of our previous reviews of IPEs for other plants.) The FSAR analysis uses 190° F as the maximum sump temperature and does not address operability of pumps in recirc with a saturated sump. The pumps could fail due to either excessive temperature or due to cavitation. Figures 1.2-8 and 1.2-13 of the FSAR indicate that the elevation drop from the top of the containment sump to the

suction of the RHR pumps is about 26 feet; Figure 6.3-2 of the FSAR indicates that the NPSHR for the RHR pumps at high flow is on the order of 25 feet. Thus, it is not obvious that the RHR pumps can operate with a saturated sump, although this may be the case.

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

The other PRA results available for comparison with the Watts Bar IPE are the Sequoyah NUREG-4550 study and the McGuire IPE. Because of differences in the reporting of dominant failures, the results of these studies will be listed separately.

Sequoyah - NUREG/CR-4550:
Total Core Damage Frequency: 1E-4/yr
Dominant Contributors:

	% - Contribution
LOCA (1)	>50%
Seal LOCA (2)	31%
Station Blackout	5%

McGuire - IPE:
Total Core Damage Frequency: 4E-5/yr
Dominant Contributors:

	% - Contribution
LOCA (1)	>44%
Seal LOCA (2)	45%
Transients	20%

Watts Bar - IPE:
Total Core Damage Frequency: 3.3E-4/yr
Dominant Contributors (3):

	%-Contribution
Support System Faults	44%
Loss of Offsite Power	17%
Loss of Coolant Accidents (1)	12%
Transients	11%
ATWS	10%
Internal Floods	4%
SGTR	2%

Notes:
(1) Dominated by small LOCA loss of recirculation.
(2) Dominated by support system faults resulting in loss of RCP seal integrity - loss of injection.
(3) 70% of Watts Bar's Core Damage Frequency involves loss of RCP seal integrity.

The results from these three studies are similar in that all three plants have a high contribution to core damage from failures of the RCP seals and failure of ECCS recirculation.

The Watts Bar IPE presented an importance analysis of plant hardware on total CDF. As can be seen below, the systems involved in maintaining RCP seal integrity are dominant:

System	Percentage Importance
Component Cooling Water System	37
Essential Raw Cooling Water System	21
Offsite and Common AC Power	17
Reactor Coolant System; i.e., LOCAs	17
Shutdown Boards	15
Auxiliary Feedwater System	13
Emergency Diesel Generators	12
Reactor Trip System	11
Main Feedwater & Condensate Systems	10
125V DC Power System	8

The following paragraphs indicate the level of discussion and type of information presented for the top ten accident sequences. The discussions indicate that the sequences were expanded to identify the dominant contributors. Sequences 1 and 3 are really the same. They differ only in that in sequence 1 the operators attempt to refill the RWST using the containment spray system and in sequence 3 they do not; however, the containment spray system fails due to loss of lube oil cooling so it is unavailable for refilling the RWST in both sequences. Sequences 4 and 7 are really the same sequence for the same reason.

Sequence 1 - (1.28E-5/yr). The highest frequency core damage sequence begins with a total loss of both trains of the component cooling system (CCS). This system failure results in a loss of cooling to the reactor coolant pump (RCP) thermal barrier heat exchangers and to the upper and lower motor bearings. The loss of motor bearing

cooling means that after a period of time with continued pump operation, loss of effective lubrication leads to pump vibration and eventual failure of the bearings. The vibration of the motor and shaft would be transmitted to the RCP seals, which are assumed to fail after a period of time, estimated to be at least 2 minutes, and realistically, 10 minutes is assumed.

The operators are directed by procedures to trip the reactor and the RCPs in the event of no CCS flow to the RCP oil coolers; however, there could be some hesitation in this sequence since their action will result in tripping the unit, and the operators would first try to recover from the loss of all CCS cooling.

Failure to trip the RCPs prior to seal damage is assumed to lead to a small loss of coolant accident (LOCA). The loss of all CCS cooling means that oil cooling for the centrifugal charging pumps and the safety injection pumps is also lost. However, the operators successfully stop the initially running train A charging pump, align essential raw cooling water (ERCW) cooling to its oil coolers, and restart the pump. Charging pump A then provides high pressure injection to make up for the loss of inventory through the damaged RCP seals.

The residual heat removal (RHR) heat exchangers, however, are unavailable for recirculation from the sump due to the loss of CCS cooling. The RHR pumps would also be stopped once recirculation from the sump is required due to the loss of pump seal cooling. Continued operation of the RHR pumps taking suction from the hot water in the sump, without seal cooling, could lead to a LOCA outside containment as a result of an eventual failure of the RHR pump seals.

Therefore, recirculation from the containment sump is unavailable. The loss of sump recirculation leads to eventual core damage due to the loss of inventory control.

The containment spray pumps are also unavailable in this first sequence for recirculation from the containment sump due to the loss of lube oil cooling. The spray pumps may run for a period of time in the injec n mode, while taking suction from the cooler refueling water storage tank (RWST) ir itory. Operation of the spray pumps in the injection mode would reduce containment pressure but would also shorten the time to empty the RWST.

Sequence 2 - (1.26E-5/yr). This sequence is initiated by a total loss of all ERCW cooling, i.e., inadequate ERCW flow to both trains A and B. The frequency for loss of all ERCW is derived from combinations of failures involving pumps failing to run, failing to start, and being in maintenance, and of check valves failing to reseat on demand, which leads to a diversion of flow from other operable pumps.

Loss of all ERCW, which provides the safety grade ultimate heat sink at Watts Bar, means that CCS cooling is also unavailable. This means that all emergency core cooling system (ECCS) pumps and all RCP seal cooling would be lost. Consequently, core damage is assumed to result eventually due to an RCP seal LOCA without any injection from the RWST. Auxiliary feedwater (AFW) is still available because the turbine-driven AFW pump ventilation is not dependent on ERCW or CCS.

Sequence 3 - (1.13E-5/yr). This sequence is similar to Sequence 1. The only difference is that in this sequence, the operators also fail to attempt to provide make up to RWST for continued high pressure injection. Since containment spray is unavailable to provide make up to the RWST from the containment sump anyway, the added operator action has no additional impact. No credit for makeup to the RWST via the VCT is assumed as an alternative to recirculation from the sump.

Sequence 4 - (7.68E-6/yr). This sequence is a variation on the top sequence. In this case, there is also a total loss of all CCS cooling. Unlike the top-ranked sequence, in this sequence, the operators successfully trip the RCPs in time to prevent RCP seal damage

due to motor bearing failure. However, the operators do not align ERCW cooling to the centrifugal charging pump for continued RCP seal injection. Therefore, long-term damage to the RCP seals occurs due to loss of all seal cooling. Since none of the high pressure injection pumps have lube oil cooling, core damage results from the seal LOCA without high pressure injection. Again, for the sequence, both the RHR and containment spray pumps are also unavailable.

Sequence 5 - (7.20E-6/yr). This sequence is initiated by a loss of offsite power. The Unit 1 onsite diesel generators 1A-A and 1B-B both fail to start. The turbine-driven AFW pump operates successfully so that secondary heat removal is successful. However, the loss of all shutdown power on Unit 1 and a consequential loss of all RCP seal cooling lead to an RCP seal LOCA. Electric power from offsite or from onsite is not recovered before the loss of RCS inventory out the failed RCP seals lead to core uncovery. Core damage then occurs. In the current evaluation of these sequence, no credit was given for aligning the fifth, or C-S, diesel generator to Unit 1, or for powering pumps at unit 1 with unit 2 diesel generators.

Sequence 6 - (6.93E-6/yr). This sequence is initiated by a small LOCA that is assumed to occur at the RCP seals of one pump. The plant trips, and a safety injection signal is generated. Both the charging and the safety injection pumps actuate to provide RCS inventory control at high pressure. Containment spray pumps come on in response to a high-high containment pressure condition. Automatic swapover of RHR suction of the containment sump is successful, but the operators are postulated in this sequence to fail to align for high pressure recirculation, i.e., the discharge of the RHR pumps is not aligned to the suction of the charging or safety injection pumps. Core damage results because of the loss of inventory control while in recirculation.

In the Level 2 analysis, this sequence was evaluated using the Modular Accident Analysis Program (MAAP) (IPE Reference 3.4-1) thermal-hydraulic analysis program. Given a successful cooldown by the operators (i.e., as directed by their post-LOCA cooldown

procedures), MAAP shows that core damage would not occur due to failure of high pressure recirculation, provided that the RHR pumps operate for low pressure recirculation from the sump. The accumulator inventory keeps the core covered while the RCS cool downs and depressurized sufficiently for low pressure recirculation. This sequence is assumed to be arrested within the vessel for the Level 2 analysis.

Sequence 7 - (6.7 3E-6/yr). This sequence is similar to Sequence 4. The only difference is that in this sequence, the operators also fail to attempt to provide makeup to the RWST for continued high pressure injection. Since containment spray is unavailable to provide makeup to the RWST from the containment sump anyway, this added operator failure has no additional impact. No credit for makeup to the RWST via the volume control tank (VCT) is assumed as an alternative to recirculation from the sump.

Sequence 8 - (5.97E-6yr). This sequence is initiated by a loss of train A of Component Cooling Water with continued operation of Train B. Train A of CCS provides the cooling for the RCP thermal barrier heat exchangers and for the RCP motor bearing coolers. Similar to the top sequence, the operators fail to trip the RCPs in time to prevent seal damage due to pump vibration. A small LOCA is assumed to develop. Since Train B of CCS is still available in this sequence, Train B of the charging and the safety injection pumps operate for RCS inventory control until the RWST empties. Containment spray pump Train B actuates when containment pressure reaches the high-high containment pressure setpoint and then continues to provide containment heat removal in the recirculation mode. The RHR pump B fails independently. Loss of both RHR pump trains results in a failure of core cooling during recirculation.

Sequence 9 - (5.26E-6/yr). This sequence is similar to Sequence 6. The only difference is that in this sequence, the operators also fail to attempt to provide makeup to the RWST for continued high pressure injection. Since containment spray in unavailable to provide makeup to the RWST from the containment sump anyway, this added operator failure has

no additional impact. No credit for makeup to the RWST via the VCT is assumed as an alternative to recirculation from the sump.

Sequence 10 - (3.44E-6/yr). This sequence is initiated by a total loss of all (i.e., both trains) component cooling water. The operators successfully trip the RCPs in time to prevent early RCP seal failure due to pump vibration. Charging pump A (i.e., the only pump that can currently be aligned to ERCW for backup lube oil cooling independent of CCS) fails independently. Therefore, due to loss of CCS, none of the charging or safety injection leads to eventual seal failure. Seal failure results in a small LOCA. The failure of all high pressure injection pumps then leads to core damage with inventory loss through the RCP seals. Neither the RHR pumps nor the containment spray pumps are available for long-term containment heat removal due to the loss of CCS.

In summary, the Watts Bar IPE has presented the most probable core damage sequences and has identified the dominant contributors to each sequence.

II.1.2.7 Front-End and Back-End Interfaces

The Level 1/Level 2 interfacing was accomplished through a set of Plant Damage States (PDS). A PDS matrix was used to delineate all possible PDS's. The following specific items were considered during the Watts Bar PDS binning:

1. RCS Pressure at Core Damage
   - P< 200 psia
   - 200 psia < P < 600 psia
   - 600 psia < P < 2,000 psia
   - P < 2,000 psia

2. Steam Generator Cooling
   - Yes
   - No

3. RWST Injection Into the Containment
   - Yes
   - No

4. Containment Isolation and Bypass Status
   - Containment Isolated and not Bypassed
   - Containment not isolated or failed prior to core damage; leak area less than the equivalent of approximately 3" dia.
   - Small containment bypass (SGTR)
   - Large containment bypass (V-Sequence)

5. Containment Spray Operation - Injection and Recirculation
   - Operable
   - Not operable

6. Containment Heat Removal
   - Yes
   - No

7. Ice Condenser
   - Available
   - Not Available

8. Hydrogen Control
   - Available
   - No Available

II.1.2.8 Multi-Unit Considerations

The Watts Bar plant is a two-unit site with Unit 1 being the lead unit. Section 1 of the IPE submittal states: "The PRA models are developed for Unit 2 systems that are shared with Unit 1. The results are applicable to Unit 1 only; Unit 2 is still under construction."

II.1.3 Review of IPE Quantitative Process

II.1.3.1 Quantification of the Impact of Integrated Systems and Component Failures Quantification Process

The quantification of core damage sequences for the Watts Bar IPE involved two main steps:

- Sequence Assembly
- Sequence Quantification

Sequence assembly requires the linking of:

1. The initiating events that have been identified in the analysis.
2. Support system event trees that model the functional relationship among support systems.
3. Frontline event trees that model the functional relationship among operator actions, equipment, and instrumentation in frontline systems that are important to risk.

Sequence quantification requires the assignment of a split fraction value to each branch in each linked event tree. A split fraction value is the conditional frequency of failure for a given event tree top event. When the appropriate information is provided, the RISKMAN software quantifies each sequence through the tree one at a time. The frequency of each sequence is computed by multiplying by the product of the branch frequencies along that sequence path. To account for intersystem dependencies, the branching frequencies are quantified dependent on the status of preceding top events in

the tree. When sequences are quantified in RISKMAN, the mean values of the split fraction, initiating event, and human error rate distributions are used. Finally, the frequencies of the core damage sequences are combined to yield an overall core damage frequency. The value obtained is 3.3 x 10-4/yr.

Uncertainty Analysis

The Watts Bar IPE uncertainty analysis consisted of the propagation of uncertainties from the basic events through the final core damage sequences. The uncertainty in the overall core damage frequency is computed by the Importance Sequence Module of RISKMAN. The total core damage frequency is presented as a distribution with a mean of 3.3 x 10-4/yr, and a 95th percentile of 7.0 x 10-4/yr.

II.1.3.2 Fault Tree Component Failure Data

Plant-Specific Data

The Watts Bar Plant has yet to operate - no plant-specific data is available.

Sources of Generic Failure and Rational for Their Use

The primary source of generic initiating event frequency data for the Watts Bar IPE was from:

- PLG, Inc. "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," PLG-0500, Vol - 6, PWR Initiators, proprietary, August 1989.

A spot check of this data was made against NUREG/CR-2815; PSA Procedures Guide. This check indicated reasonable consistency.

The primary source for generic component failure data is:

- PLG, Inc., "Database for Probabilistic Risk Assessment for Light Water Nuclear Power Plants," proprietary, PLG-0500, July 1989.

The parameter estimates in this database are presented in terms of probability distributions which characterize the uncertainty of the data. The mean values are used as point estimates in the models. The PLG database has already been reviewed for the NRC (NUREG/CR-5606). The reviewers found that the database was extensive and "state of the art". A comparison of the PLG database with the data contained in NUREG-2815, Appendix C indicates reasonable consistency.

Common Cause Data

The Watts Bar IPE, used the Multiple Greek Model (MGL) to generate common cause failure rates. The primary generic data source for estimating the component common cause parameters is the PLG generic database. The generic parameters were reinterpreted as per NUREG/CR-4780, to develop a Watts Bar specific database.

II.1.4 Review of IPE Approach to Reducing CDF

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

The Watts Bar IPE states that "a vulnerability may exist if the mean core damage frequency exceeds 5×10^{-4} per-year. Several PWR plants evaluated using similar PRA data and methods have been reported to the NRC total core damage frequencies in the range at 5×10^{-5} to 5×10^{-4} per reactor-year.

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. That section reviews noteworthy Watts Bar safety features and discusses potential improvements. However, no commitments are made for plant enhancements based on the IPE results.

II.1.5 Review of Licensee's Evaluation of DHR Function

The Watts Bar IPE claims the following modes of decay heat removal:

1. Auxiliary Feedwater - Steam Pump or Atmospheric Dump Valves. The Watts Bar Plant has three AFW pumps - 2 motor driven pumps, 1 turbine driven pump. These pumps can draw water from either the CST or the ERCW system.

2. Main Feedwater - Steam Dump or Atmospheric Dump Valves. The Watts Bar Plant has three MFW pumps - 2 motor driven pumps, 1 turbine driven pump. The MFW pumps take suction from the suction of the condensate booster pumps which in turn take suction from the condenser hotwell. The MFW system is isolated on a reactor trip but can be recovered.

3. RHR System - Closed Loop & Sump Cooling

4. Feed and Bleed Cooling

The Watts Bar IPE takes credit for feed and bleed cooling with one charging or one SI pump and with both PORVs open.

It should be noted that the IPE addressed loss of DHR. However, the IPE should have considered depressurization and use of condensate for cooling if feedwater is lost. The IPE assumption is conservative, but other IPEs have credited this option for DHR.

It should be noted that the Watts Bar IPE states on page 3.1.2-9 that if the RHR pumps are not stopped after 100 minutes of operation on miniflow recirculation, they will fail. The IPE claims that if a Phase B containment isolation condition also occurs and containment spray actuates successfully, switchover to recirculation would occur before the pumps overheat. Therefore the possibility of pump overheat is neglected. However:

1. On page 3.1.2-6 the IPE states that the operators will reset the Phase B signal and stop the containment spray pumps once containment pressure falls below the Phase B setpoint.

2 On page 3.1.2-55 the IPE states that for small LOCA sequences, containment spray operation would empty the RWST in just a "couple of hours."

The licensee provided clarifying information related to this issue which indicates that this assumption has a small impact on the overall results of the IPE. Should this not be the case, it may be important to:

- determine the time to recirculation switchover after a small break LOCA with continuous containment spray operations.
- determine the time to recirculation switchover after a small break LOCA with the core spray system turned off and high pressure injection throttled to match the break flow rate.
- determine the time to RHR closed-loop entry conditions if the operators successfully align a refill source to the RWST, stop the containment spray pumps, and throttle injection to match the break flow rate.

II.1.6 USI and GSIs

The Watts Bar IPE submittal does not address USIs & GSIs, other than loss of DHR.

III. OVERALL EVALUATION AND CONCLUSION

In conclusion we note that: the finalization of the Technical Specifications, the possibility of plant changes to resolve Generic Issue 23: Reactor Coolant Pump Seal Failure, and the possibility of plant changes to comply with the Station Blackout Rule, may influence the representation of the plant. It may be beneficial to the licensee that the IPE be updated when the operational design of the plant becomes finalized. In addition, the analysts should:

- Correct the logic errors in the Sequence definitions, as described in Section II.1.2.6 of this report.
- Perform Thermal Hydraulic timing studies both to evaluate the likelihood of success of key actions outlined in plant procedures (e.g., steam generator depressurization for low pressure condensate injection) and to confirm LOCA success criteria.
- Perform a sensitivity analysis on the assumption that a loss of motor bearing oil cooling will lead to a seal LOCA.

IV. IPE EVALUATION AND DATA SUMMARY SHEETS

This section includes the data sheets related to the front-end portion of the Watts Bar IPE. The format of this Appendix follows that provided by the NRC in our task statement. The section numbers are according to the NUREG-1335 standard Table of Contents.

2.4 Information Assembly

Per NUREG-1335, the IPE should provide a list of PRA studies of their plant or other similar plants that the IPE team has reviewed along with or list of the important insights derived from this review. It appears that the following PRA analysis were reviewed by the IPE team.

1. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment," prepared for Pacific Gas and Electric PLG-0637, July 1988.
2. PLG, Inc., "South Texas Project Probabilistic Safety Assessment," Houston Lighting and Power Company, PLG-0700, Vols. 1 and 2, April 1989.
3. Sandia National Laboratories, "Analysis of Core Damage Frequency: Sequoyah Unit 1 Internal Events," prepared or the US Nuclear Regulatory Commission, NUREG/CR 4550.
4. PLG, Inc., Westinghouse Electric Corporation, and Fauske and Associates, Inc. "Indian Point Probabilistic Safety Study," prepared for the Power Authority of the state of New York and Consolidated Edison Company of New York, Inc., March, 1982.
5. Westinghouse Electric Corporation, "Sequoyah Nuclear Plant Individual Plant Examination," WCAP-11769, November 10, 1988.

The Watts Bar submittal did not include a list of plants of similar design. The submittal did, however, discuss differences is the PRA results between the Watts Bar IPE and the results from the NUREG/CR-4550 study of Sequoyah. In addition to Watts Bar and Sequoyah, other 4-loop Westinghouse Pressurized Water Reactors with Ice Condenser Containments are:

McGuire (Duke Power)
Catawba (Duke Power)

3.1.1 Initiating Events

The Watts Bar IPE presents the following initiating events:

Watts Bar Initiating Event Categories	
Initiating Event	
Group	**Category**
Loss of Coolant	1. Excessive LOCA (reactor vessel failure) 2. Large LOCA (> 6-inch diameter) 3. Medium LOCA (≥ 2 to ≤ 6-inch diameter) 4. Small LOCA (nonisolable) 5. Small LOCA (isolable) 6. Steam Generator Tube Rupture 7. Interfacing Systems LOCA - RHR Injection Path 8. Interfacing Systems LOCA - RHR Suction Path
Transients	9. Reactor Trips 10. Core Power Excursion 11. Turbine Trip 12. Inadvertent Safety Injection 13. Total Loss of All Main Feedwater 14. Partial Loss of Main Feedwater 15. Loss of Condenser Vacuum 16. Excessive Feedwater 17. Inadvertent Closure of On MSIV 18. Inadvertent Closure of all MSIVs 19. Loss of Primary Flow 20. Steam Line Break Outside Containment 21. Steam Line Break Inside Containment 22. Inadvertent Opening of Main Steam Relief Valves
Loss of Support Initiating Events	23. Loss of Offsite Power 24. Loss of 1A-A, 6.9-kV Shutdown Board 25. Loss of 1B-B, 6.9-kV Shutdown Board 26. Loss of 1-I Vital AC Instrument Board 27. Loss of 1-II Vital AC instrument Board 28. Loss of 1-III Vital AC Instrument Board 29. Loss of 1-IV Vital AC Instrument Board 30. Loss of Vital Battery Board I 31. Loss of Vital Battery Board II 32. Total Loss of CCS 33. Los of CCS Train A 34. Total Loss of ERCW 35. Loss of ERCW Train A 36. Loss of ERCW Train B

Internal Flooding Events	37. Turbine Building Flood - Loss of Feedwater, Condenser, and Station Air 38. ERCW Strainer Room A Flood - Loss of All Four "A" Pumps and Headers 39. ERCW Strainer Room B Flood - Loss of All Four "B" Pumps and Headers 40. ERCW Flood in Auxiliary Building for 30 Minutes - RHR and Containment Spray Unavailable 41. CST Drained to Auxiliary Building - CST, RHR, Containment Spray, and one AFW Pump Unavailable 42. RWST Drained to Auxiliary Building - RWST, RHR, and Containment Spray Unavailable

The I.E. contribution to CDF was presented as follows:

Support System Faults	Mean Annual CDF (Per reactor-year)	Percentage of Total
Support System Faults	1.5×10^{-4}	44
Loss of Offsite Power	5.5×10^{-5}	17
Loss of Coolant Accidents	4.1×10^{-5}	12
Transients	3.4×10^{-5}	11
ATWS	3.2×10^{-5}	10
Internal Floods	1.4×10^{-5}	4
SGTR	7.8×10^{-6}	2
ISLOCA	4.1×10^{-6}	<1

The Licensee also stated the following:

- 70% of CDF at WATTS Bar involves RCP seal failure.
- The total loss of component cooling water initiator accounts for 17.4% of CDF. The loss of CCS Train A initiation accounts for 10% of CDF.

3.1.2 Front-Line Event Tree Review

Basis for Success Criteria

The major plant functions modeled in the IPE are reactor criticality control, early core heat removal, reactor coolant system integrity, containment pressure system integrity,

containment pressure suppression, late core heat removal, and containment atmospheric heat removal. The success criteria for these major plant safety functions were determined through engineering knowledge of the plant and a review of the following documents.

1. Bertucio, R.C., et al., "Analysis of Core Damage Frequency: Sequoyah, Unit 1 Internal Events," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-4550, Vol. 5, Rev. 1, April 1990.
2. EG&G Idaho, Inc., "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-3862, May 1985.
3. U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risk in U.S. Nuclear Power Plants," WASH-1400, NUREG-75/014, 1975.
4. American Nuclear Society and Institute of Electrical and Electronic Engineers, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments of Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute, NUREG/CR-2300, April 1983.
5. Pickard, Lowe and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, Inc., "Indian Point Probabilistic Safety Study," prepared for the Power Authority of the State of New York and Consolidated Edison Company of New York, Inc., March 1982.
6. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," PLG-0500, Vol. 6, PWR Initiators, proprietary, August 1989.
7. Stillwell, D.W., "Memorandum on the Importance of Watts Bar Unit 1 HVAC Systems in the IPE," May 1992.
8. Tennessee Valley Authority, "Watts Bar Unit 1 Final Safety Analysis Report (FSAR)," April 1992.

RCP Success Criteria

No RCP success criteria is discussed in the IPE.

HVAC Assumptions:

Electrical Power System HVAC

There are three ventilation systems modeled; the shutdown board room ventilation system, 480v board room ventilation, and 480v. Transformer room ventilation system. Based on plant specific calculations from Stillwell, D.W., "Memorandum on the Importance of Watts Bar Unit 1 HVAC Systems in the IPE," May 1992 the following recovery times are allowed for specified areas prior to equipment damage from excessive temperatures.

6.9kV Train A Shutdown Board Room Ventilation Systems

Ventilation for the Unit 1 and Unit 2 Train A 6.9kV shutdown boards are supplied by air handling units A-A and C-B. To prevent equipment damage, the restoration of ventilation to the shutdown boards must be completed within 12 hours.

6.9kV Train B Shutdown Board Room Ventilation System

The Unit 1 and Unit 2 Train B 6.9kV shutdown boards are cooled by Air Handling Units B-A and D-B. To prevent equipment damage, the restoration of ventilation to the shutdown boards must be completed within 12 hours.

480V Board Room Ventilation

The Unit 1 Train A and Train B 480V shutdown boards are supplied by Air Handling Units A-A and C-B. The restoration of ventilation to the shutdown boards must be completed within 12 hours.

480V Shutdown Transformer Rooms 1A and 1B Ventilation

The Unit 1 480V shutdown transformer rooms require active ventilation. Four fans draw from room 1A. Three of these fans are powered from 480V C and Ventilation Board 1A1-A. The fourth fan is powered from 480V Common Board A. Three fans draw from Room

1B. All three fans are supplied power by 480V C&A Ventilation Board 1B1-B. The establishment of portable ventilation to the rooms affected by the failed ventilation system must be completed within 10 hours for Room 1B, and 5 hours for Room 1A.

CCS/AFW Pump Area Coolers

The component cooling water and motor driven auxiliary feedwater pumps require active ventilation. Success required that two of the four ESF area coolers operate on demand for 24 hours or, that both non-ESF Elevation 713' air handling units and associated auxiliary building cooling water operate for 24 hours after a plant trip. Failure of these ventilation systems is assumed to fail the operating CCS and motor-driven AFW pumps. No credit appears to be given for recovery actions.

Charging Pump Room HVAC

No dependencies on HVAC for the charging pumps are discussed in detail. However, in the transient event tree description, the room coolers are stated to be modeled with the pumps. No possible recovery actions are discussed.

RHR Pump Room HVAC

No dependencies on HVAC for the RHR pumps are discussed in detail. However, in the transient event tree description, the room coolers are stated to be modeled with the pumps. No possible recovery actions are discussed.

SIS Pump Room HVAC

Operation of the SIS pumps requires operation of the pump room coolers. No recovery actions are discussed.

CCS Pump Room HVAC

No dependencies on HVAC for the CSS pumps are discussed in detail. However, in the transient event tree description, the room coolers are stated to be modeled with the pumps. No possible recovery actions are discussed.

Diesel Generator Room HVAC

No dependencies on HVAC for the diesel generators are discussed in detail. However, in the transient event tree, the room coolers are stated to be modeled with the pumps. No possible recovery actions are discussed.

3.1.3 Special Event Tree Review

Three different LOCA events are postulated in the Watts Bar IPE:

1. RCP Seal LOCAs are postulated to occur under conditions in which CCS flow to the bearing oil coolers has failed, the RCPs are still running, and the operators fail to trip the pumps on a high bearing oil temperature alarm. The RCP seals are assumed damaged after 10 minutes from excess pump vibration. This event is developed as a small LOCA.
2. Failure of both seal injection and thermal barrier cooling will result in a small LOCA.
3. To allow for the recovery of AC power and subsequently coolant injection prior to core damage for station blackout sequence, the IPE uses a probability leak rate model to calculate the time to core uncovery after a SBO induced RCP seal LOCA. The pump seal leak model was based on the RCP seal LOCA study (NUREG/CR-5116-Vol-1), "Results of Expert Opinion Elicitation on Internal Event Front-End Issues for NUREG-1150: Expert Panel," Sandia 88-0642, April 1988) for Westinghouse RCPs with the old style O-rings that exist in the Watts Bar Plant.

Seal LOCA Flow Rates (GPM) with and without Primary Depressurization							
Probability	Cumulative Probability	Flow Rate (in GPM) versus Time after Station Blackout					
		0-1.0 (hours)	1.0-1.5 (hours)	1.5-2.5 (hours)	2.5-3.5 (hours)	4.5-5.5 (hours)	5.5+ (hours)
0.2712	.2712	84	84	84	84	84	84
0.0151	.2863	84	84	84	244	244	244

0.0161	.3024	84	84	244	244	244	244
0.0181	.3205	84	244	244	244	244	244
0.0120	.3325	84	244	433	433	433	433
0.0059	.3384	84	244	433	433	480	698
0.1120	.4505	84	244	1,000	1,000	1,000	1,000
0.0136	.4640	84	480	1,000	1,000	1,000	1,000
0.5302	.9942	84	1,000	1,000	1,000	1,000	1,000
0.0016	.9958	84	1,230	1,230	1,230	1,230	1,230
0.0042	1.0000	84	1,920	1,920	1,920	1,920	1,920

3.1.4 Support System Event Tree Review

The event tree methodology employed was the large event tree/small fault tree (LET/SFT) approach. The support system and front-end event trees are directly coupled whereby accident sequences are traced from the initiating events through the support system event tree, then through the front-end event trees.

The consultant employed was Pickard, Lowe, and Garrick, Inc.

3.2 System Analysis (Dependency Matrix-BNL)

3.2.2 Fault Trees

Fault trees were stated to have been developed for the following systems. They were not provided in the Tier I submittal documentation.

Electric Power System
Essential Raw Cooling Water System
Component Cooling System
Plant Compressed Air System

Chemical and Volume Control System

Condensate and Feedwater System

Engineered Safety Features Actuation Systems

Reactor Protection System

Auxiliary Feedwater System

Main Steam System

Residual Heat Removal

Safety Injection System

RCP Seal Injection and Thermal Barrier Cooling

Pressurizer Power-Operated Relief Valves and Safety Valves

Steam Generator Isolation

Containment Spray System

Containment Systems

3.2.3 System Dependencies

Plant Unique System Dependencies

No plant-unique system dependencies were identified, and none were noted in the submittal.

Important Plant Asymmetries

The RCP thermal barrier coolers are supplied from Train A CCS only.

3.3.1 List of Generic Data

Three forms of generic data were used in the Watts Bar IPE.

Component Failure Rates

Component Maintenance Frequency and Duration

Internally Caused Initiating Event Frequencies

. ne Watts Bar database, for all forms of data, was developed primarily based on the PLG proprietary database. (Pickard, Lowe and Garrick, Inc., "Database for Probabilistic Risk Assessment for Light Water Nuclear Power Plants," proprietary PLG-0500, July 1985).

3.3.2 Plant Specific Data and Analysis

Source of Plant-Specific Data

Since Watts Bar has yet to operate, no plant-specific data was available.

3.3.4 Common Cause Failure Analysis

Technique Used to Treat Common Cause Failures

Multiple Greek Letter (MGL) with reinterpretation of common cause events using the method described in NUREG/CR-4780 (Moslesh, A. et al.,), "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," prepared for Electric Power Research Institute and U.S. Nuclear Regulatory Commission, NUREG/CR-4780, Vols I and II, 1988.)

Level of Treatment

Component Groups

Most Significant Common Cause Failures

Because of the PLG methodology of combining basics events into split fractions, it was difficult to determine the most dominant common cause events in the top core damage sequences. However, significant common cause failures can be inferred from the information provided in the submittal.

1. Common Cause Failures Involving Component Cooling Water System Equipment
2. Common Cause Failure Involving Essential Raw Cooling Water System Components
3. Common Cause Failures Involving Emergency Diesel Generators

Source of Common Cause Data

PLG-0508, Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants

3.3.5 Quantification of Unavailability of Systems and Functions

Systems or Components With Noted Unusually High or Low Unavailability

Since Watts Bar has yet to operate, no plant-specific unavailabilities are available.

Source of Test and Maintenance Unavailabilities and Repair Rates

PLG-0500, Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants.

3.3.7 Quantification of Sequence Frequencies

Codes Employed in the Quantification Process

Quantification was performed using the RISKMAN Code.

Uncertainty Analysis or Sensitivity Analysis Performed

Importance analysis was performed. The importance analysis ranked the percentage contribution to the total core damage frequency made by all sequences grouped by common failed split fractions. Sensitivity studies were performed to analyze the effects of plant changes on CDF.

3.3.8 Internal Flooding

Methodology

The Watts Bar flooding analysis methodology included the following steps:

1. Plant Familiarization - Key plant design information was reviewed.
2. Flood Experience Review - Flood data collected from Nuclear Power Experience (S.M. Stoller Corporation, Nuclear Power Experience, updated

monthly) was reviewed to ensure familiarity with actual flood events, their locations within the plant, and their causes.

3. Evaluation of Flood Sources - Using the plant design information and a general knowledge of plant layout, major flood sources and their locations were identified.
4. Evaluation of Plant Locations - Using plant design information such as arrangement drawings, internal flood studies, and information from the evaluation of flood sources, the buildings most important to risk were identified.
5. Plant Walk Through - A walk-through was performed to collect additional information and to confirm previous documentation and judgements on flood sources, their potential impact, propagation paths, and detection.
6. Scenario Quantification - Based on the above steps, scenarios were postulated, evaluated, and quantified for use as initiating events. The effect on plant systems from the initiator is also evaluated.
7. Risk Model - To develop flooding core damage sequences, the flood scenarios were included as initiating events to the transient event tree.

Contribution of Internal Flooding to Core Damage

The contribution to core damage from internal flooding was calculated to be 1.419×10^{-5}/yr or 4.3% of the total core damage, frequency. This contribution is dominated by floods in an essential raw cooling water strainer room followed by random failures which would defeat the other ERCW train. The loss of ERCW cooling will lead to RCP seal failure.

Critical Internal Flood Areas

- ERCW Strainer Rooms

Most Critical Flood Sources

- ERCW from the Tennessee River

3.4.1 Application of Screening Criteria

Form of Truncation, Probability Frequency of cutset size.

Only the 26 plant damage states with a frequency of 1×10^{-7} or greater per reactor-year were carried through to the back-end analysis.

Definition of Core Damage

No definition of core damage was provided.

Dominant Accident Sequences

(BNL Data Entry)

Dominant Contributors to Core Damage

The dominant contributors listed by functional groups, and their percent contribution are:

Support System Faults	44%	
Loss of Offsite Power	17%	
LOCAs	12%	
Transients	11%	
ATWS	10%	
Internal Floods	4%	
SGTR		2%
IS LOCA	<1%	

Recovery Actions (Sequence Level)

The Watts Bar IPE included the following recovery actions.

(a) Cross-connections between buses/trains

- Align ERCW header 1A to Charging Pump A after loss of component cooling.
- Align ERCW header 2A to CCS Heat Exchangers A, given loss of B train ERCW.

- Recover cooling to diesel generator by aligning the opposite ERCW header to the diesel.
- Align the C-S pump to the A CCS Heat Exchanger.

(b) Restoration/repair of secondary side cooling

- Restore AFW flow, given loss of control air.
- Locally transfer steam supply to TDAFP, given station blackout and loss of steam generator #1.
- Start Turbine Driven Pump given it failed to start due to control or signal failures.

(c) Alternate Emergency Power Sources

- Restore Operation of emergency diesel generator or restore offsite power.
- Switch to spare battery charger, given operating charger fails.

(d) Other

- Recover from an automatic swapover failure.
- Identify and Isolate an ERCW intake piping in line AIA-A break in the strainer room.
- Identify and isolate an ERCW intake line break in the pump room of ERCW intake.
- Identify and isolate a break in the condensate storage tank discharge piping. Given break can be isolated.
- Identify and isolate ERCW header flood in auxiliary building.
- Cooldown with AFW and steam generator PORVs, given total loss of ERCW, RSW available to cool air compressor.
- Cooldown with AFW and S/O PORVs, given total loss of ERCW, RSW not available to cool air compressors.

- Restore ventilation to the 480V board room 1BB (2BB), given loss of room supply F9N.
- Establish portable ventilation to the shutdown bcard transformer room.

3.4.2 Vulnerability Screening

Importance on Relative Ranking Provided ?

Importance evaluations were performed relative to initiating events, event tree top events, split fractions, and human actions.

Licensee's Definition of Vulnerability

The licensee states that a vulnerability may exist if the mean core damage frequency exceeds 5×10^{-4}.

Vulnerabilities (Identify Specifically DHR Related Vulnerabilities)

The submittal states that no vulnerabilities were identified.

Plant fixes in response to identified vulnerabilities and change in core damage frequency if known

The licensee states that no DHR vulnerabilities exist.

Consideration of plant life extension in proposed plant modification (Y/N)

No considerations of plant life extension was evident in the Watts Bar IPE.

3.4.3 Decay Heat Removal

Methods of Removing Decay Heat

The decay heat removal during the first 24 hours following a plant trip is accomplished by the following means at Watts Bar.

- Main Condenser
- RHR System

- Steam Generator Atmospheric Dump Valves.

Ability of the Plant to Feed and Bleed

The Watts Bar plant can feed and bleed using one charging or safety injection pump and two PORVs. Each PORV is designed to relieve 210,000 pounds per hour of saturated steam at 2,265 psia.

Credit for Feed & Bleed

The Watts Bar IPE takes credit for Feed and Bleed Cooling.

Credit for Recovery of power conversion system?

The Watts Bar IPE takes credit for restoring main feedwater but not for reopening the MSIVs.

Steam Generator Dryout Time

One hour it assumed in the analysis to be available before the steam generators dry out (page 3.1.2-36).

Main Feedwater Trip on Reactor Trip

For normal plant response to a simple plant trip, the main feedwater (MFW) regulating valves close when reactor coolant system (RCS) T_{Avg} drops below the required setpoint. The MFW pumps also trip on this feedwater isolation signal.

Unique Front-End System Features

Important Unique Features:

1. The availability of diesel generators on the opposite unit means that power for steam generator level instrumentation will still be available even for an extended station blackout on the unit analyzed.
2. No procedures exist that allow utilization of the fifth diesel generator.

3. The RHR and CSS utilize different containment sump isolation valves.
4. The CSS system can be used to refill the RWST from the containment sump.
5. The ECCW System can be used to provide backup cooling to charging pump A after loss of CCS.
6. ERCW is automatically aligned as a water source to the AFW pumps on low AFW suction header pressure.
7. The condensate storage tank has a relatively large capacity.
8. There are eight ERCW pumps for the two Watts Bar units compared to four pumps, at other two-unit stations.
9. A standby electric-driven main feedwater (MFW) pump is available.
10. Both steam generator level instrumentation power and turbine-driven AFW pump control power is supplied from the opposite unit.

6. Plant Improvements and Unique Safety Features

Important Insights Including Unique Safety Features.

The following insights were discussed in the IPE.

1. Enhancements to the operator training and procedures for responding to failures of support systems could potentially be beneficial, with emphasis on anticipating problems and coping.
2. Ventilation has been conservatively modeled in this study. Area ventilation is provided to the motor-driven AFW pumps and the CCS pumps from multiple systems serving the plant elevation where these pumps are located. Beyond design basis concurrent failures of the available unit 1 ventilation are assumed to impact the long-term availability of the AFW and CCS. An evaluation of the CCS/AFW area cooling requirements could be

performed which could reduce this interdependence by creating natural convection and availability of other coolers at this plant elevation.

3. In the event of a loss of ERCW, which would eventually lead to a loss of CCS cooling, additional guidance on the relationship of CCS to ERCW and the desirability of eliminating CCS loads to extend the time of suitable CCS temperatures is a potential consideration for evaluation. This could be accomplished by revising AOI-13, "Loss of ERCW," to alert the operators to shed CCS loads prior to CCS heatup.

4. During a loss of all AC, the steam generator power-operated relief valves (PORV) are to be locally operated to depressurize the steam generators, thereby cooling down the RCS. The addition of provisions for remote operation of these valves could potentially be beneficial due to the high area temperatures that may be encountered.

5. In the event of a loss of CCS cooling to the charging pumps, the time available for operation of the pumps would be limited by the loss of lube oil heat exchanger cooling. To extend the time that is available to protect the pumps, consideration could be given to increasing the oil capacity.

6. Losses of RCP seal cooling could potentially be reduced if the RCP thermal barrier cooling dependence on component cooling water, which is required for the charging pumps that provide RCP seal injection, could be eliminated.

7. Currently, ventilation for the 480V board room that contains the unit vital inverters is provided by one train of ventilation. The current models rely substantially on recovery actions by the operators. Consideration could be given to providing two trains. This condition has been previously documented and is being resolved by the Watts Bar corrective action program.

8. From a severe accident point of view, one potential change, for consideration, could be the delaying of spray operations relative to the Phase B, condition. Currently, containment sprays actuate immediately in response to a Phase B condition, and air return fans (ARF) actuate after a

10-minute delay. This is currently a requirement of the design basis LOCA where switchover to containment spray recirculation occurs prior to ice melt, thereby limiting pressure increases below containment design pressure. Modular Accident Analysis Program (MAAP, Reference 6-2) analyses of representative core damage sequences indicate that actuation of the containment sprays while ice remains in the ice condenser has little impact on severe accident containment performance and may be detrimental in that operation of the sprays rapidly depletes the inventory of the RWST making its contents unavailable for vessel injection. Since many scenarios have successful injection but failure at recirculation, the rapid depletion of the RWST due to spray operation accelerates the time to core damage. Therefore, an evaluation balancing the scenario accident versus design basis requirements could be made.

9. For addressing the loss of CCS train A, consideration should be given to revising AOI-115, Loss of Component Cooling Water to facilitate stopping the RCPs on loss of CCS train A to minimize the potential for RCP seal damage due to pump bearing failure.

10. In the event of a total loss of CCS, clearer guidance should be considered that directs the operator to cool down the RCS prior to seal damage.

11. In the event of a loss of offsite power followed by the failure of both shutdown boards on one unit, the procedures would be enhanced by adding the guidance to align the C-S diesel generator (i.e., the fifth diesel generator) to one of the shutdown buses.

12. Provide connections for both centrifugal charging pumps on both units to the ERCW system for lube oil cooling in the event of a loss of CCS cooling to the associated pump.

ATTACHMENT 3

WATTS BAR UNIT 1 INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)