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STEP 1 TECHNICAL EVALUATION REPORT OF THE WATTS BAR UNIT 1 INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

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TABLE OF CONTENTS

1. INTRODUCTION

This technical evaluation report (TER) documents the results of the SCIENTECH Step 1 Review of the Watts Bar Unit 1 (WB1) Individual Plant Examination (IPE) Back-End submittal [1]. This technical evaluation report complies with the requirements of the U.S. Nuclear Regulatory Commission contractor task order for Step 1 reviews, and adopts the NRC Step 1 review objectives, which include the following:

- To determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335
- To assess the strengths and the weaknesses of the IPE submittal
- To pose a preliminary list of questions about the IPE submittal, based on this limited Step 1 review
- To complete the IPE Evaluation Data Summary Sheet.

In Section 2 of the TER, we summarize our findings and briefly describe the WB1 IPE submittal as it pertains to the work requirements outlined in the contractor task order. Each portion of Section 2.1 corresponds to a specific work requirement. In Section 2.2, we set out our assessment of the WB1 submittal strengths and weaknesses, and, in Section 2.3, we submit our questions, comments, and requests for more information to the IPE submittal authors. In Section 3, we present our evaluation of the WB1 IPE overall, as well as our conclusions, based on the Step 1 review. Appended to this report is an evaluation summary sheet completed on the WB1 IPE.

2. CONTRACTOR REVIEW FINDINGS

2.1 Review and Identification of IPE Insights

This section is structured in accordance with Task Order Subtask 1.

2.1.1 General Review of IPE Back-End Analytical Process

2.1.1.1 Completeness

The Watts Bar Unit 1 Individual Plant Examination (IPE) Back-End submittal is essentially complete with respect to the level of detail requested in NUREG-1335. The IPE submittal meets the NRC sequence selection screening criteria described in Generic Letter 88-20.

2.1.1.2 Description, Justification, and Consistency

The IPE methodology used is described clearly and its selection is justified. The approach followed is consistent with Generic Letter GL 88-20, Appendix 1.

2.1.1.3 Process Used for IPE

Section 1.3, page 1-2 of the submittal, notes that the WB1 IPE used

> a Level 2 PRA in terms of the key characteristics of radioactive material releases that could result from the scenarios identified. The results currently reported are for a Level 2 PRA, as defined in the IEEE/ANS "PRA Procedures Guide" [2] and Appendix A to Generic Letter No. 88-20.

The containment analyses of WB1 IPE were performed using NUREG-1150 study of Sequoyah plant. After reviewing 114 top events of the Sequoyah accident progression event tree, the IPE analysts selected 30 top events for the CET, which they used to quantify the WB1 Level 2 analysis. The review results are summarized in Table 4.5-1, pages 4.5-12 through -26 of the submittal.

2.1.1.4 Peer Review of IPE

Section 5 of the IPE submittal describes the independent, in house peer review of the WB1 plant. The submittal notes that the Tennessee Valley Authority (TVA) conducted training sessions on PRA for the site personnel. The TVA presented the accident sequences, insights, and recommendations of the IPE to various site organizations.

Table 5.1 lists the Level 2 review participants. The peer review team did not find any major deficiency and team comments received response on an ongoing basis. We cannot judge the extent of the peer review, however, because the submittal reports no results.

Consultants used in this IPE were from PLG, GKA, and EQE.

2.1.2 Containment Analysis/Characterization

2.1.2.1 Front-end and Back-end Dependencies

The WB1 IPE considered the front-end and back-end dependencies, using 66 plant damage states (PDSs). Section 4.3.1, page 4.3-1 of the submittal, notes the following:

> Before PDSs are defined, the analyst must identify plant conditions, systems, and features that can have a significant impact on the potential course of an accident. Once these are identified, a table is constructed to display the potential combinations of the PDS characteristics that are physically possible, and to assign an identifier to each of these combinations. The table that results from this process is referred to as a PDS "matrix." The matrix is then reviewed by Level 1 analysts to ensure compatibility with the plant model and that appropriate dependencies are taken into account. The matrix is revised as necessary until it satisfies the requirements of Level 1 and Level 2 analysts.

The WB1 PDS matrix is shown in Figure 4.3-1, page 4.3-8 of the submittal.

The same section notes the following containment system availability considerations included in the PDS matrix (page 4.3-4):

- The state of the containment (intact or failed) at the time when core damage starts, i.e., when the containment event tree is entered. This distinction includes containment isolation failure and interfacing system LOCA considerations.

- The availability of engineered safety features, such as containment sprays, the ice condenser, and heat removal, to cool the containment atmosphere and fission product removal before and after failure of the reactor vessel.

- The availability of hydrogen control systems.

The treatment of system dependencies between the front-end and the back-end appear to be adequately treated. All the active containment systems (and their support systems) are included in the front-end analysis. This information is brought into the back-end analysis through the definition of the plant damage states. Thus, the auditing and understanding the dependencies is fairly straight forward.

2.1.2.2 Sequence with Significant Probability

As can be seen in Table 4.6.2, page 4.6-13 of the submittal, the WB1 IPE meets or exceeds the IPE reporting criteria specified in GL 88-20.

2.1.2.3 Failure Modes and Timing

The containment failure characterization is discussed in Section 4.4 of the submittal. The following modes were analyzed for the containment failure, defined as an incipient leakage, for containment metal temperatures from room temperature to 800°F:

- Cylinder hoop failure
- Dome membrane failure
- Equipment hatch buckling
- Containment anchor bolt failure
- Personnel hatch bulkhead flexure
- Baseslab failure
- Pipe penetration failure.

The capacities of the above failure modes are given in Table 4.4-1 (page 4.4-6). The baseslab failure mode had the lowest median pressure capacity; the equipment hatch buckling and dome membrane failure modes had the next lowest. These three failure modes were addressed in the IPE containment event tree (CET) quantification in which the dome membrane failure and equipment hatch buckling were calculated using the finite elements method in the reference cited. [3] The baseslab failure, which was not analyzed in Reference 3, was calculated by assuming it was a uniform, axisymmetric flat plate. The median failure pressure of baseslab at room temperature was 95 psig.

The baseslab failure is below ground while the other failures of interest are above ground. When containment failures are addressed in the containment event trees, they are divided into large containment failures and small breaches in the containment. If the failure is determined to be a small breach, the analysis is continued until, possibly, a large breach occurs later in the accident progression.

2.1.2.4 Containment Isolation Failure

Containment isolation failure appears to have been analyzed using two top events performed in the WB1 IPE: Top Event 1, "Containment not bypassed prior to core damage (BY)" (page 4.8-37) and Top Event 10, "No containment failure prior to vessel breach (CI)." Section 4.8.2, page 4.8-19 of the submittal, notes that pre-existing leaks, both large and small, were considered in the Level 1 model and treated as an unisolated containment. Containment isolation failures can be large, (e.g., purge lines) and small, (e.g., seal return lines).

Of the top 66 PDSs listed in Table 4.6-1, pages 4.6-12 and -13 of the submittal report, 32 PDSs involved containment isolation failure, with a total frequency of 8.7E-6 per reactor

year, or 2.6 percent of the total core damage frequency estimated for the WB1 plant. This is a relatively high value for isolation failure.

2.1.2.5 System/Human Response

System/human interaction appears to be analyzed using two top events of the CET: Top Event 3, "Core damage arrested prior to vessel breach," and Top Event 7, "No induced RCS hot leg or Surge line failure." Table 1 gives the split fractions used in Top Event 3 (Table 4.8-2, pages 4.8-37 and -38):

Table 1. Split Fractions Used for Top Event 3, Core Damage Arrested Prior to Vessel Breach

Failure Fraction	Basis
0.00	Sequences KPDS LNIYA (bleed and feed successful with one PORV) are not predicted to go to core damage.
0.23	Sequences in KPDS LCI that have bleed and feed with one PORV were shown by MAAP analysis to not go to core damage.
0.68	Sequences in KPDS FCI that have small LOCA with SG cooldown and LP recirculation were shown by MAAP analysis to not go to core damage.
0.77	Sequences in KPDS BCI that have medium LOCA with SG depressurization and LP recirculation were shown by MAAP analysis to not go to core damage.
1.00	Most accident sequences reflect a relatively short period of time between the uncovering of the top of the active fuel and vessel breach.
1.00	Default value.

It is not clear how much of the Top Event 3 success (for KPDSs LNIYA, LCI, and FCI) is attributed to human reliability (for "feed and bleed," for example) and how much is attributed to a reassessment of the thermal hydraulic success criteria. Likewise, the role of the operator, if any, in the success of Top Event 7 is unclear. A question is suggested in Section 2.3, to clarify this.

2.1.2.6 Radionuclide Release Characterization

Section 4.9 of the submittal describes the radionuclide release characterization. The following are the WB1 source term characterization parameters and their attributes (page 4.9-5):

- Containment bypass (yes or no)
- Time of containment failure (pre-VB, early, late, or none)
- Size of containment failure or bypass (large break or small leak)
- Location of containment failure of bypass (to environment, to auxiliary building, or to ground)

- Containment spray operation following VB (injection and recirculation, injection only, or none)
- Ex-vessel debris cooling (cooled, scrubbed, or no)
- Ice condenser functionality for the RCS release (intact, partially bypassed, or bypassed)
- Air return fan operation for the RCS release (yes or no)
- RCS pressure at time of VB (high, medium, or low).

Table 4.9-1 lists why the above parameters are important and whether they are PDS or CET events.

Using the above parameters, Table 4.9-2 defines 207 release categories, which are grouped into four release categories: Groups I, II, III, and IV. After arranging these release categories in descending order of release frequency, the IPE analysts selected 10 key source term categories (KSRCs). These KSRCs are 6 for Group I (R01D1, R031F, R04, R03I, R01IF, R01); 1 for Group II (R20); and 3 for Group III (R17L, R17LU, and R17U). The source term for each of these KSRCs was calculated using MAAP and each is listed in Tables 4.9-1 through -13. Each listing includes time of release and release fractions for 12 fission product groups, which include noble gases, cesium iodide, tellurium (oxide/elemental), strontium, molybdenum, cesium hydroxide, barium, lanthanides, cerium, antimony, and uranium, and actinides.

Table 2 reproduces the major contributors to a large, early release as listed in Table 1-8, page 1-14 of the submittal.

Generic Letter 88-20 states that the following should be reported:

> [A]ny functional sequence that has a core damage frequency greater than 1×10^{-6} per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to BWR-3 or PWR-4 release categories of WASH-1400.

We could not locate in the submittal any reference to the above criterion, and it is not clear whether the IPE satisfied this reporting requirement.

Table 2. Major Contributors to Large, Early Release Frequency

Type of Event	Percentage Contribution*
SGTR (with bypasses to the environment)	55
Containment failure due to direct impingement	24
α-Mode failure of vessel/containment	14
HPME/hydrogen burns at vessel beach	5
Hydrogen burns/DDT before and after vessel breach	2
Interfacing system LOCAs	<1

*Sum of Release Category Group I and SGTR bypasses from Group II

The radionuclide release characterization is well presented and complete. For example, it is straight forward to track the KPDS contributions to the release category groups.

2.1.3 Quantitative Core Damage Estimate

2.1.3.1 Severe Accident Progression

The WB1 IPE analyzed severe accident progression using the Modular Accident Analysis Program (MAAP). [4] The version used was MAAP/PWR Revision 17.02 with the code changes as given in Appendix 4A to the submittal report.

Section 4.7.2 of the submittal describes the MAAP runs made for selected sequences of 14 KPDSs for which results are shown in Figures 4.7-1 through -59 (pages 4.7-16 through -74). Results shown are time variations of primary system pressure, containment pressure, hydrogen production, hydrogen volume fraction, oxygen volume fraction, environmental release fractions, steam generator collapsed liquid level, gas temperature, and mass of ice in the ice condenser.

The WB1 IPE used sensitivity analyses to address uncertainties in key assumptions involved in the following areas of severe accident progression (Section 4.10.6, pages 4.10-4 and -5): α-mode vessel and containment failure, debris impingement containment failure, and hydrogen burns and the importance of hydrogen igniters and air return fans.

Sensitivity studies are also discussed in Section 4.8.1.2 of the submittal. The sensitivity studies included those recommended in the Electric Power Research Institute report, "Recommended Sensitivity Analysis for an Individual Plant Examination using MAAP 3.0B."

2.1.3.2 Dominant Contributors Consistency with IPE Insights

In Table 3, we compare dominant contributors analyzed in the WB1 IPE with those analyzed in the McGuire IPE and NUREG-1150 study of Sequoyah. The predicted core damage frequency calculated for WB1 was about an order of magnitude higher than for McGuire and Sequoyah. The relative contribution to radioactive release from containment isolation failure of WB1 was an order of magnitude higher than that reported for the McGuire IPE.

The Watts Bar plant is quite similar to the Sequoyah plant in all the characteristics that are important to assessing the back-end portion of the PRA with the exception of the design pressure. For Sequoyah it is 10.8 psig, while for Watts Bar it is 15 psig. This may account for part of the difference in the early failure percentages, comparing the two plants. A detailed comparison of the two plants is displayed in Table 4.1-1, pages 4.1-10 & 11.

Another marked difference in the containment failure characteristics between Sequoyah and Watts Bar is in the relatively large "basemat melt-through" for Sequoyah. This appears to result from the relatively low probability for reactor cavity flooding for Sequoyah.

Table 3. Containment Failure as a Percentage of CDF:*
WB1 Comparison with McGuire IPE and Sequoyah NUREG-1150 Results

Containment Failure	McGuire IPE	Sequoyah/ NUREG-1150	Watts Bar 1 IPE
CDF (per reactor year)	4.1E-5	5.6E-5	3.3E-4
Early Failure	2.0	7.6	1.9
Late Failure	35	3.8	27
Basemat Melt-through	5.9	17	2.0
Bypass	2.4	5.6	2.1
Isolation Failure	0.3	na	2.6
Intact	54	64	65

* Total may be different from 100 percent because of round off errors.

2.1.3.3 Characterization of Containment Performance

The WB1 IPE characterized containment performance using containment event trees. For the WB1 CET, the analysts selected 30 top events after reviewing 111 top events of the NUREG-1150 study of Sequoyah plant. [5] As shown in Table 4.5-1 of the submittal, some top events in the Sequoyah CET were not considered for the WB1 CET because they were present in the Level 1 plant damage states. Several top events in the Sequoyah CET were combined to form WB1 CET top events. The WB1 CET top events constitute three categories: Top Events 1 through 12 represent the events before the vessel breach, Top Events 13 through 22 and 29 represent the events at or around the vessel breach, and Top Events 23 through 28 represent the events describing long-term containment behavior. Table 4 describes the 30 top events used in the WB1 CET. Figure 4.5-1 of the submittal shows the WB1 CET.

The WB1 IPE used the RISKMAN computer program to analyze the detailed logic of the CET and to propagate probabilities of accident sequences. As noted in Section 4.5.1, page 4.5-2 of the submittal, phenomenological questions were used as top events if they addressed any of the following issues:

- Definition of a safe, stable state for the debris configuration either in- or ex-vessel
- Dependencies for later top events in the CET, such as reactor coolant system (RCS) pressure at vessel breach, high-pressure melt ejection, hydrogen burns, basemat melt--through
- Containment failure events and failure mode.

The CET is well developed and the description of the Top Events and the CET quantification is well presented. It is straight forward to track the accident progressions from the PDSs through to the various release categories and containment failure characteristics.

2.1.3.4 Impact on Equipment Behavior

Equipment survivability in a severe accident environment is discussed in Section 4.1.4 of the submittal report. For operation after the vessel breach, the IPE credited only air return fans, hydrogen igniters and containment sprays, designed for LOCA containment conditions. As noted on page 4.1.4.5 of the IPE submittal

> [t]he air return fans, including the fan drive motors, backdraft dampers, and power cabling, are contained inside containment between the upper and lower compartments. Two redundant AC-powered fans are provided, each of which is capable of circulation sufficient air to satisfy the purpose of the air return fans.

Table 4. Watts Bar Unit 1 Containment Event Tree Top Event Descriptions

Number	Designator	Description
0	IE	Plant damage state
1	BY	Containment not bypassed before core damage
2	LB	No large bypass before core damage
3	CV	Core damage arrested before vessel breach
4	LS	No induced pressurizer PORV or SRV failure
5	SP	Reactor coolant pump seal cooling available
6	IS	No steam generator tube rupture
7	IP	No induced RCS hot leg or surge line failure
8	HO	No hydrogen burn before vessel breach
9	ICE	No loss of ice condenser function before vessel breach
10	HC1	No loss of containment failure before vessel breach
11	L1	No large containment failure before vessel breach
12	RP	RCS pressure at vessel breach
13	ME	No high-pressure melt ejection
14	C2	No containment failure at vessel breach
15	L2	No large containment failure at vessel breach
16	DI	No direct impingement of debris on seal table wall
17	X2	Heat removal available immediately after vessel breach
18	DBC	Debris cooled
19	HE	No early hydrogen burn
20	CE	No containment failure due to early burn
21	LE	No large containment failure due to early burn
22	XE	Containment heat removal available after early burn
23	CL	No containment failure due to late burn
24	LL	No large containment failure due to late burn
25	XLT	Long-term containment heat removal available
26	CLT	No long-term overpressurization containment failure
27	LLT	No large long-term containment failure
28	BI	No basemat penetration
29	SO	Sprays operate after large, early failure

The AC-powered hydrogen igniters are located inside the containment and are designed to survive the temperature conditions that would result from controlled hydrogen burns. Page 4.1-8 of the submittal notes that

> [t]he containment spray pumps are located in the auxiliary building and stainless steel spray nozzles are employed in the spray ring headers located inside containment, so that spray injection should not be impaired by ambient containment conditions.

The WB1 IPE identified the following parameters, which provide important information to the operators on hardware located in the containment in the aftermath of core damage (page 4.1-7 of the submittal):

- Reactor coolant system pressure
- Reactor vessel water level
- Core exit temperature
- Pressurizer water level
- Containment pressure
- Containment area radiation
- Containment hydrogen concentration
- Containment atmosphere temperature
- Steam generator secondary-side water level.

The hardware located inside the containment to monitor these parameters is designed for LOCA containment conditions.

2.1.4 Reducing Probability of Core Damage or Fission Product Release

2.1.4.1 Definition of Vulnerability

As defined in Section 3.4.3, page 3.4-6 of the submittal, "a vulnerability may exist if the mean core damage frequency exceeds 5×10^{-4} per reactor-year or the mean large early release frequency exceeds 5×10^{-5} per reactor-year." Section 6.3, page 6-4 of the submittal notes that no vulnerabilities exist for WB1. The origin of these definitions of "vulnerability" was not given.

2.1.4.2 Plant Improvements

The WB1 IPE does not recommend any plant improvements because no vulnerabilities were found to exist (See Section 6.3 of the submittal). The IPE submittal does recommend some enhancements of plant hardware and procedures for Level 1, but none for Level 2.

2.1.5 Responses to CPI Program Recommendations

Supplement No. 3 to GL 88-20 states that, as part of an IPE, the licensees with ice condenser containments might evaluate the vulnerability of hydrogen igniters to power interruptions. In response to this recommendation, the WB1 IPE submittal (Section 4.10.4, pages 4.10-2 and -3) notes that

> [f]ollowing the recovery of AC power, the operators are instructed to start the containment spray pumps if containment pressure is above the phase B condition and to determine the volumetric concentration of hydrogen. If this concentration exceeds 6 percent, the ignitors are not energized. If large concentrations of hydrogen are present, there are other sources of ignition which could trigger a hydrogen burn. Since these ignition sources may be of random nature, an ignition could occur at any time following recovery of AC power. To demonstrate that Watts Bar has no specific vulnerability to ignitor unavailability, it was conservatively assumed that all of the CDF associated with KPDS with the ignitors unavailable would result in containment failure at some time.

No other containment performance improvements were considered.

2.2 IPE Strengths and Weaknesses

2.2.1 IPE Strengths

1. The back-end portion of the submittal is well written and logically developed.

2. The submittal demonstrates extensive knowledge of severe accident progression and phenomenolgy and the impact of severe accidents on containment system response. (However, it is difficult to determine from reading the submittal how much of this knowledge resides with the utility staff as compared to the contractor personnel.)

3. The WB1 IPE used sensitivity analyses to address uncertainties in key assumptions involved in the following areas of severe accident progression (Section 4.10.6, pages 4.10-4 and -5): α-mode vessel and containment failure, debris impingement containment failure, and hydrogen burns and the importance of hydrogen igniters and air return fans. Also, the WB1 IPE analyzed the sensitivity of "core damage arrest" on the frequencies of various release category groups, for which the results are given in Table 4.10-4, page 4.10-24 of the submittal.

4. Figures 4.7-1 through -59, pages 4.7-16 through -74 of the submittal, provide detailed MAAP results.

5. The TVA educated site personnel about PRAs by conducting training sessions.

6. Appendix 4A to the IPE submittal lists the WB1 IPE changes to the MAAP/PWR code, Revision 17.02.

2.2.2 IPE Weaknesses

1. The insights gained and consideration of ways to fix vulnerabilities were focused almost exclusively in the front-end. (An exception is the consideration of delaying containment spray operation to allow for more RCS coolant injection capability; see page 6-6 of the submittal.) While attention to the front-end is very important, a more deliberate and systematic look at the back-end may be appropriate, especially considering the large uncertainties in both the front and the back-end.

3. OVERALL EVALUATION AND CONCLUSIONS

As discussed in Section 2, this IPE submittal contains a large amount of back-end information, which contributes to the resolution of severe accident vulnerability issues at Watts Bar. The back-end portion of the PRA is well written and the authors understand the back-end containment technology. The questions raised in Section 2 address some areas and issues that do not appear to be completely addressed in the IPE submittal. There appear to be no major weaknesses. Additional key points of the review are:

- The submittal did not appear to completely address all of the generic letter reporting requirements. There is only a brief discussion of potential containment performance improvements and no discussion of how the PRA met the radiological source term screening requirements.

- The results are driven, to some extent, by the assumption that induced hot leg failure of the RCS will allow for depressurization prior to vessel failure, thus reducing the probability of early containment failure. While this is consistent with NUREG 1150 results for PWRs, a more in-depth review of the Watts Bar analysis in this area may be justified.

- The recovery of coolability before core damage and vessel failure goes beyond Level 1 recovery considerations. While this might be valid, it may be desirable to perform a more detailed evaluation of recovery potential. It appears that MAAP is used for determining thermal hydraulic success criteria.

4. REFERENCES

1. Tennessee Valley Authority, "Watts Bar Unit 1 Individual Plant Examination Report," September 1992.

2. American Nuclear Society and Institute of Electrical and Electronics Engineers, "PRA Procedures Guide: A Guide to the Performance of the 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.

3. J. Jung, "Ultimate Strength Analysis of the Watts Bar, Main Yankee and Bellefonte Containments," NUREG/ CR-1 3724, July 1984.

4. R. E. Henry and M. G. Plys, "MAAP-3.0B - Modular Accident Analysis Program for LWR Power Plants," Electric Power Research Institute, EPRI NP-7071-CCML, November 1990.

5. Sandia National Laboratories, "Evaluation of Severe Accident Risks: Sequoyah, Unit 1," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-4551, SAND86-1390, Vol. 5, Rev. 1, Part 2, December 1990.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

PWR Back-end Facts

Plant Name

Watts Bar Unit 1

Containment Type

PWR ice condenser

Unique Containment Features

None found

Unique Vessel Features

None found

Number of Plant Damage States/Key Plant Damage States

66/17

Ultimate Containment Failure Pressure

95 psig

Additional Radionuclide Transport and Retention Structures

Ice condenser and auxiliary building effectiveness credited

Conditional Probability That the Containment Is Not Isolated

0.026

APPENDIX (continued)

IPE EVALUATION AND DATA SUMMARY SHEET

Important Insights, Including Unique Safety Features

- Procedures to ensure containment isolation
- Three relief valves in the residual heat removal system with large relief capacity that vent to the containment
- Four separate pumps that can take suction from the containment sump during recirculation, (i.e., two RHR pumps and two containment spray pumps).

Implemented Plant Improvements

None

APPENDIX (continued)

IPE EVALUATION AND DATA SUMMARY SHEET

C-Matrix

KPDS	Frequency per yr	Early Failure	Late Failure	Intact
ENI	1.4E-4	0.01	0.23	0.75
FCI	9.0E-5	0.01	0.03	0.96
FNI	4.3E-5	0.01	0.86	0.14
LCI	1.5E-5	0.04	0.01	0.96
GNI*	8.3E-6	0.31	0.96	
BCI	5.7E-6	0.01	0.77	0.23
EIB	5.5E-6	1.00		
ENS	4.8E-6	0.98		
HGI	3.1E-6	0.20	0.81	
ENB	2.7E-6	0.99		
EGI	2.3E-6	0.01	1.00	
KNS	2.2E-6	0.98		
KNI	2.2E-6	0.16	0.82	
FGI	2.1E-6	0.01	0.86	0.14
HNI	2.0E-6	0.08	0.95	
HCI	1.6E-6	0.01	0.05	0.94
LNI	1.4E-6	0.11	0.54	0.38

* Containment failure probabilities do not add to 1.00.

ATTACHMENT 4

WATTS BAR UNIT 1 INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)