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BROWNS FERRY UNIT 2 TECHNICAL EVALUATION REPORT ON THE INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

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1. INTRODUCTION

This technical evaluation report (TER) documents the results of the SCIENTECH Step 1 Review of the Browns Ferry Unit 2 (BF2) 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 BF2 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 BF2 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 BF2 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 BF2 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 Browns Ferry Unit 2 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 clearly described and its selection is justified. The approach followed is consistent with Generic Letter 88-20, Appendix 1.

2.1.1.3 Process Used for IPE

In Section 1.3, page 1-2, of the submittal, it is stated with respect to the Level 2 analysis that "The results reported here are based on the methods that conform to the NRC guidelines [Generic Letter 88-20, Appendix 1] and IEEE/ANS 'PRA Procedures Guide' [NUREG/CR-2300]." The Level 1 model quantification led to identification of 30 plant damage states (PDSs) with a frequency of $1x10^{-12}$ per year or greater. For the Level 2 analysis, these PDSs were condensed into a set of nine key plant damage states (KPDSs) for which containment event trees (CETs) were developed. Representative sequences were selected for each KPDS. The MAAP computer code was used to calculate severe accident event timing and containment loads for each of the representative sequences.

2.1.1.4 Peer Review of IPE

Overall, the participation of Browns Ferry Unit 2 in the independent peer review of Level 2 analysis appears to be limited. In any event, the record of participation appears to be contradictory. According to Table 5-2 of the Browns Ferry IPE submittal, only two individuals from Browns Ferry participated in the review of Level 2 Containment Analysis: one from Site Engineering and the other from Site Civil Engineering. On the other hand, however, Table 5-3 lists as participants in the Level 2 analysis independent peer review members of the Browns Ferry staff from Site Engineering, Operations, Technical Support, and KAS. The record of peer review performance is incomplete as well. Table 5-3 refers to completion of the following actions: "Review and incorporate Level 2 comments. Concurrence with Level 2 model." However, neither these comments nor points of concurrence can be located in the submittal.

2.1.2 Containment Analysis/Characterization

2.1.2.1 Front-end to Back-end Dependencies

No discussion of front-end to back-end dependencies was found in the back-end portion of the submittal report. SCIENTECH reviewed and agreed with the front-end to back-end dependency issues raised by the NRC contractor, SEA. SEA reviewed the front-end portion of the BF2 IPE.

2.1.2.2 Sequence with Significant Probability

In Section 4.6.1 of the submittal, the following is stated with respect to CET evaluating sequences with significant probability:

[T]he nine KPDSs selected collectively account for 99.1% of the total core damage frequency. Also note that the key PDSs with containment failure or bypass collectively represent more than 95% of the total frequency in this category and include all such 'containment failure' PDSs with frequencies greater than 10^{-7} per year. Therefore, all Level 2 selection criteria are satisfied.

2.1.2.3 Failure Modes and Timing

The BF2 containment structural evaluation and failure characterization are discussed in Section 4.4 of the IPE submittal report. EQE Engineering Consultants analyzed containment overpressure capacity and the following failure modes were identified:

- Drywell sphere-to-cylinder knuckle
- Drywell cylinder
- Drywell closure flange
- Torus
- Drywell-to-torus vent line
- Numerous hatches
- Personnel air lock
- Electrical penetrations.

The limiting failure modes were found to be failures of the drywell knuckle and drywell closure flange for which failure pressures and uncertainties were calculated for temperatures of 200, 300, 400, 600, and 800 degrees Fahrenheit. As shown in Table 4.4-2 of the submittal, the median value of drywell closure flange leakage pressure dropped from 202 psig at 200°F to 165 psig at 800°F. For drywell knuckle and drywell closure flange failure modes, the resultant cumulative failure probability distribution was calculated and results are shown in Figure 4.4-7. These results were used in MAAP calculations to obtain failure timing.

2.1.2.4 Containment Isolation Failure

Containment isolation failure is addressed in Section 4.5.2.2, page 4.5-5 of the submittal, Top Event 3 - Containment Intact before Vessel Breach (I1), as follows:

> [Top Event 3] addresses preexisting containment leak paths due to isolation failures, or the possibility that containment failure can occur prior to core damage (as defined by the plant damage state) or any induced containment failures that could occur prior to vessel breach.

The KPDS MKC, which resulted from containment isolation failure or containment failure prior to core damage within a few hours of event initiation, had a conditional core damage frequency of 0.82 percent.

2.1.2.5 System/Human Response

System/human response was analyzed using CET Top Event 7 - Drywell spray initiated prior to vessel breach (DS), and Top Event 14 - Emergency crew vents containment in core damage scenarios (DV). (See Section 4.5.2, pages 4.5-6 and 4.5-8 of the submittal.) For the PDS PIH, which consisted of 59.8 percent of total core damage frequency, operator failure for DS was 3 percent. (See Table 4.8-1, page 4.8-6.) As noted in Tables 4.8-1 through -5, no credit was taken for operator venting (i.e., DVF = 1.0) for five top KPDSs that consisted of 96.3 percent total core damage frequency. For KPDSs PIH, MIA, and NIH: "[It was assumed] for extended blackout cases that dirty venting would not occur due to electric power and plant air vent valve dependencies"; for KPDSs OIA and PID, "In the absence of hard vents no manual venting is deemed possible."

2.1.2.6 Radionuclide Release Characterization

In Section 4.9 of the submittal, radionuclide release categories and source-term characterizations are discussed. As shown in Figure 4.9-1 of the submittal, a BF2 source-term event tree (STET) was developed with six top events as given below. As noted in the same section, "[The STET] is used to develop the CET binning logic for assigning the appropriate release categories for each CET sequence."

- Vessel breach or drywell spray
- Pressure at vessel breach: low/high
- Containment status: intact, vented, or subject to failure timing
- Controlled or gross failure
- Torus scrubbing
- Reactor building mitigations.

Using these top events, the BF2 IPE analysts identified 64 release categories, which were condensed into 10 key release categories (KRCs) as follows: EARLYSCRUB, EARLYHIDWFAIL, EARLYATWS, EARLYBYPASS, LATESCRUB, PROMPTTW, DELAYEDTW, LATENOSCRUB, NRELNOVB, and NRELNOCF. Table 4.9-1 of the submittal summarizes the results of the condensation process for various release categories. Of the 10 KRCs, two were identified as having no significant offsite releases (NRELNOVB and NRELNOCF). In Section 4.7.2 of the submittal, noble gas and CsI release fractions are listed for representative KRCs. These fractions were calculated using MAAP. A summary of release frequency results listed according to release category groups appear in Table 1.8, page 1-13 of the submittal.

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

> any 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.

The BF2 IPE submittal appears to have met this reporting requirement, although there is no specific reference to it. (See Section 4.6.1 of the submittal).

2.1.3 Quantitative Core Damage Estimate

2.1.3.1 Severe Accident Progression

In Section 4.7 of the IPE submittal, the analysis is discussed of severe accident progression performed using MAAP 3.0B computer code. As shown in Figure 4.7-1, page 4.7-49 of the submittal, the node and junction diagram was used to perform this analysis at BF2. As noted in Section 4.7.1.4, a MAAP parameter file was developed, consisting of Browns Ferry data on the reactor coolant system, primary containment, and secondary containment.

Results of MAAP calculations are presented as summaries, tables, and figures. Result summaries of 15 calculation cases appear on pages 4.7-8 through 4.7-28 of the submittal. For each case, there is a brief description of the scenario, timing for key events, significant results, and a discussion of them. Timing is stated for the following events:

- Battery depletion
- Core uncovery
- Onset of melting
- Vessel failure
- Drywell shell failure
- Drywell knuckle failure
- Reactor building equipment hatch cover failure
- Hydrogen burn in reactor building
- Refuel floor blowout.

Significant results are given on the following:

- Vessel pressure at vessel failure
- Peak drywell pressure
- CsI release fraction to reactor building
- CsI release fraction to environment.

Table 4.7.2, page 4.7-36 of the submittal, lists (for 29 sequences) vessel pressure at vessel failure, key event times, containment conditions at time of containment failure, and CsI distribution. However, as noted in Item 1.3 of Section 2.2.2 of this report, Table 4.7.2 may be in error because the text that describes the table does not reflect the contents of the table. Figures 4.7-2 through -18 of the submittal list (for nine sequences) results of thermal-hydraulic response and noble gas and cesium iodide source terms.

As noted in Section 4.7.3 of the submittal, phenomenological uncertainties of severe accidents were addressed by performing MAAP sensitivity analysis runs. Twenty-five such calculations were performed in the following areas:

- Core melt progression
- In-vessel hydrogen generation
- RPV pressure at vessel failure
- Late CsI revaporization from RPV
- Debris spread in containment
- Amount of debris retained in RPV
- Ex-vessel debris coolability
- Shell failure
- Containment failure location
- Containment failure area
- Reactor building effectiveness

2.1.3.2 Dominant Contributors: Consistency with IPE Insights

In Table 1, below, we compare dominant contributors to BF2 containment failure with those contributors identified during individual plant examinations performed at the Fitzpatrick and Oyster Creek plants, and with the NUREG/CR-1150 results obtained at Peach Bottom.

Table 1. Containment Failure as a Percentage of CDF: BF2 Comparison with Fitzpatrick IPE, Oyster Creek IPE, and Peach Bottom NUREG-1150 Results

Containment Failure	Fitzpatrick IPE	Oyster Creek IPE	Peach Bottom/ NUREG-1150	Browns Ferry IPE
CDF (per year)	$1.9x10^{-6}$	$3.2.10^{-6}$	$4.5x10^{-6}$	$4.8x10^{-5}$
Early Failure	60.4	16.4	55.7	46
Bypass	na	7.3	na	na
Late Failure	26.0	26.4	16.0	26
Intact	2.5	0	18.0	3
No Vessel Breach	11.1	50.4	10.0	25

2.1.3.3 Characterization of Containment Performance

The containment performance observed during the BF2 IPE was characterized using containment event trees (CETs). The top events used to form the BF2 CETs are described in Section 4.5.2 of the submittal and are listed in Table 2 of this report. Each CET chronologically models core degradation, vessel failure, containment behavior, and reactor building behavior. Top Event 0 is the entry state from the front end (i.e., a KPDS). Top Events 1 through 6 could occur from the time core damage began until vessel failure seemed imminent (these events were: in-vessel breaching of core debris, safety valve failing, containment remaining intact before vessel breach, containment leaking, suppressing of pool bypass, and in-vessel and drywell spray recovery occurring).

Table 2. Browns Ferry Unit 2 CET Top Event Descriptions

Event Number	Top Event Designator	Top Event Description
CET Entry State		
0	IE	Representative key plant damage state
Events Prior to Vessel Branch		
1	IQ	In-vessel quenching of core debris
2	VS	Safety relief valve(s) (SRVs) do not stick open prior to vessel break in high-pressure melt scenarios
3	I1	Containment intact before vessel breach
4	L1	Small leak area if I1 fails
5	S1	Suppression pool not bypassed before vessel breach
6	IR	Degraded core recovered in vessel
7	DS	Drywell spray initiated prior to vessel breach
Events During or Shortly After Vessel Breach		
8	I2	Containment intact after vessel breach
9	L2	Small leak area if containment fails in Top Event 8 (I2)
10	LF	No fission products released into reactor building due to drywell shell failure
11	WD	Continued water supply to the core debris on the drywell floor after vessel breach

Table 2. Browns Ferry Unit 2 CET Top Event Descriptions (Continued)		
Long-Term Containment Events		
12	S3	Suppression pool not bypassed late
13	CC	Corium debris or drywell floor is coolable, resulting in little core-concrete interaction
14	DV	Emergency crew vents containment in core damage scenarios
15	I3	Containment intact late
16	L3	Small leak area if containment fails in Top Event 15 (I3)
Events Pertaining to Reactor Building Effectiveness		
17	HB	No hydrogen burn in reactor building
18	BE	Reactor building effectiveness in reducing offsite radiological releases
19	IL	No high-temperature-induced, long-term drywell failure

When considering Top Event 7, the BF2 IPE analysts asked when to initiate the drywell spray to flood the drywell floor and thereby reduce the likelihood of drywell shell failure by liner melt-through. In Top Events 8 through 11, phenomena related to early containment failure were analyzed. For purposes of the BF2 IPE, "Early" was defined as a 4-hour time interval following vessel breach. The phenomena that were analyzed included blowdowns, DCH effects, ex-vessel fuel-coolant interactions, and drywell shell failure due to thermal attack. In considering Top Events 12 through 16, long-term containment responses were analyzed during a 4- to 36-hour time interval following vessel breach. Top Events 12 through 16 included actions that might prevent containment failure (e.g., cooling a debris bed and removing containment heat or venting) as well as circumstances that would reduce the consequences of failure (e.g., ones that would cause a suppression pool not to be bypassed, and that would mitigate the magnitude of a leak). In considering Top Events 17 through 19, phenomena affecting the reactor building integrity and decontamination were analyzed. Split fraction documentation and logic of top events for the five top KPDSs, PIH, OIA, MIA, PID, and NIH, respectively, are given in Tables 4.8-1 through -5.

Many of the top events addressed containment behavior. The containment loading was calculated using the MAAP computer code. MAAP thermal-hydraulic results for selected sequences are shown in Figures 4.7-2, -4, -6, -8, -10, -12, -14, and -16. Each figure consists of two pressure transients (PPS - vessel and PWW - wetwell) and two temperature transients. One temperature transient is for drywell gas temperature (TGDW), but the other (TWSP) could not be identified.

Drywell shell failure were due to corium thermal attack was analyzed in Top Event 10, "No fission products released into reactor building due to drywell shell failure (LF)." As shown in Table 4.8-1 for the split fraction for Top Event DS, in a case of high vessel pressure with a dry cavity and with a wet cavity failure rates were 0.99 and 0.01, respectively; in cases of low pressure, the rates were 0.64 and 0.001. Thus the liner failure rate for a case of high vessel pressure is higher than a corresponding low pressure case. An explanation for this behavior could not be located in the submittal.

2.1.3.4 Impact on Equipment Behavior

Equipment survivability in an accident environment is addressed in Section 4.1.4, page 4.1-5 of the submittal. Exposure to degraded core environments is described with regard to the following pieces of equipment:

- Safety relief valves
- Electrical penetrations
- Equipment and personnel access hatches
- Drywell/torus vacuum breakers
- Core spray system
- Standby gas treatment system
- Condensate system.

Table 4.1-2 of the submittal indicates the locations of the above pieces of equipment and their functions, both of which are important to know in order to mitigate degraded core accidents. However, the behavior of the BF2 safety relief valves is not clear from the description in Section 4.1.4.3, which concludes that

> The most limiting environment to which the SRVs, solenoids (which are qualified for high temperature and steam environments), and power cables will be exposed directly is a degraded core environment since they are located in the drywell.

The behavior of the other equipment was adequately described for purposes of containment analysis.

2.1.4 Reducing Probability of Core Damage or Fission Product Release

2.1.4.1 Definition of Vulnerability

As noted in Section 6.3 of the submittal, no plant vulnerabilities were identified for BF2. According to Section 3.4.3, page 3.4-5 of the submittal, a vulnerability may exist if the mean core damage frequency exceeds $5x10^{-4}$ per reactor-year or if the mean large, early release frequency exceeds $5x10^{-5}$ per reactor-year.

2.1.4.2 Plant Improvements

As noted in Section 6.3, page 6-3 of the submittal, "No plant vulnerabilities were identified for BF2." Therefore, no potential enhancements were identified to address vulnerabilities. However, in Section 6.2, the prime beneficial features of BF2 were identified as they relate to Procedures/Operator Actions, Plant Hardware, and Primary Containment Operation.

- Procedures/Operator Actions
  - Integrating the responses associated with reactor primary containment and the secondary containment.
  - Delineating alternative actions, such as the use of alternate injection sources.
- Plant Hardware
  - Structural designs at Browns Ferry Unit 2 limit the impact of flooding events. For example, floods in a single RHRSW/EECW pump cell cannot propagate to other cells. Certain intake structure floods can propagate to the turbine building, but cannot propagate to the reactor building. Turbine building floods do not propagate to the reactor building.
  - The primary containment atmosphere dilution system can be aligned to provide a backup to the drywell control air system. Also, plant control air can be cross-tied to drywell control air. However, no credit was taken for these capabilities in the present analysis.
  - The drywell control air system is configured so that air from the air reserves is available to maintain the system pressure for relief valve operation. This is in addition to the accumulators connected to six of the relief valves.
- Primary Containment Operation
  - Flooding the containment prior to vessel breach substantially reduces the likelihood of drywell shell failure due to thermal attack by the core debris released to the drywell floor following vessel breach. Flooding also reduces the extent of molten corium-concrete interaction, reducing the noncondensible gases released to the containment.

2.1.5 Responses to CPI Program Recommendations

Generic Letter No. 88-20, Supplement No. 1, reiterates the following recommendations made by the Containment Performance Improvement Program (CPI) pertaining to the Mark-I containments:

- Create alternate water supply for drywell spray/vessel injection
- Enhance reactor pressure vessel depressurization system reliability
- Implement emergency procedures and training.

No response to the CPI recommendations appears to have been made in the BF2 submittal. In Section 4.10.5, however, the possibility of a sensitivity evaluation is considered to assess the alignment of the diesel-driven firewater pump to the drywell spray line in extended station blackout scenarios.

2.2 IPE Strengths and Weaknesses

2.2.1 IPE Strengths

1. Plant damage states and release categories have been conservatively condensed to a number that can be audited: 30 PDSs were reduced to 9 KPDSs and 64 release categories were reduced to 10 key release categories. The condensation process is explained clearly.

2. A sensitivity evaluation was used to assess the alignment of the diesel-driven firewater pump to the drywell spray line in extended station blackout scenarios.

3. Equipment survivability in an accident environment is addressed well.

4. Figure 4.7-1 gives the MAPP volume and junction model used for Browns Ferry Unit 2 secondary containment.

5. Phenomenological uncertainties were addressed by performing a sensitivity analysis, and in Section 4.7.3.2, each of the major uncertainties was discussed: in-vessel hydrogen generation; RPV pressure at vessel failure; late CSI vaporization; debris spread in containment; amount of debris in RPV; debris coolability; shell failure; containment failure location; containment failure size; and reactor building effectiveness.

6. The basis for severe accident progression and containment response is well presented.

7. Containment failure modes were analyzed well.

8. The plant is described in sufficient detail.

2.2.2 IPE Weaknesses

1. The BF2 IPE submittal report was poorly edited:

   1.1 No list of nomenclature can be found.

   1.2 Pages 4.4-3 and 4.4-4 appear twice, each from a different version with different contents, (i.e., once from SECT44.BFN.8/31/92 and again from SECT44.BFN.9/1/92).

   1.3 The description of Table 4.7-2 on page 4.7-6 of the submittal does not reflect the actual contents of Table 4.7-2:

   The next six columns describe the conditions of the primary containment at the time of its failure, the drywell failure area assumed in the evaluation, and whether post-vessel breach water was available to the drywell floor. The next three columns indicate the in-pedestal concrete ablation depth (feet), and the amount of in-vessel and ex-vessel hydrogen produced (pounds) at the problem end time (typically 36 hours after event initiation). The last three columns show the cesium iodide mass fractions remaining in the reactor building and released into the environment up to the problem end time.

2. No response to CPI recommendations appears to have been made.

3. Independent in-house peer review performed for the BF2 IPE appears to be limited.

4. Front-end to back-end dependencies are not explained in the submittal.

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 Browns Ferry. A large segment of the back-end portions of the IPE submittal is well written and directed to addressing Generic Letter 88-20 issues. However, there appear to be some weaknesses in the submittal, as set out in Section 2 of this report. In summary, our concerns about the submittal follow:

We believe that more attention could have been directed to the causes of early failure, the role of the operator in mitigating the consequences and ways of reducing the probability and/or consequences of the early failures. This is in light of the fact that with the core damage frequency is about an order of magnitude higher than other BWRs and the conditional probability of early failure is about 50 percent.

4. REFERENCES

1. Tennessee Valley Authority, "Browns Ferry Unit 2 Individual Plant Examination Report," September 1992.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

BWR Back-end Facts

Plant Name

Browns Ferry Unit 2

Containment Type

Mark I

Unique Containment Features

None found

Unique Vessel Features

None found

Number of Plant Damage States/Key Plant Damage States

30/9

Ultimate Containment Failure Pressure

202 psig

Additional Radionuclide Transport And Retention Structures

Reactor building and suppression pool retentions credited

Conditional Probability That The Containment Is Not Isolated

0.0082 (This value includes the conditional probability that the containment is failed prior to core damage within a few hours of event initiation)

Important Insights, Including Unique Safety Features

Secondary containment was effective in reducing the offsite release of CsI.

Liner failure due to corium thermal attack was the dominant early containment failure mode.

APPENDIX (continued)

IPE EVALUATION AND DATA SUMMARY SHEET

Important Insights, Including Unique Safety Features (continued)

Diesel-driven firewater pump was effective in providing water to the drywell spray.

Implemented Plant Improvements

None

C-Matrix*

KPDS	Frequency per year	Early Failure	Late Failure	Bypass	Intact
PIH	2.9E-5	0.52	0.30		0.21
OIA	4.9E-6	0.64	0.35		
MIA	4.7E-6				1.0
PID	4.3E-6	0.21			0.79
NIH	3.8E-6	0.41	0.24		
NLF**	9.7E-7		1.0		
MKC***	4.0E-7	0.99			
OJA	1.4E-7			0.99	
NJA	4.7E-8			1.0	

* Conditional failure probabilities for some KPDSs do not add to 1.0.

** Combines all late failure PDS frequencies (NLF+PLF+OLF+OLC+MLC).

*** Combines MKC and OKC PDS frequencies.

BROWNS FERRY NUCLEAR PLANT UNIT 2

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

HUMAN RELIABILITY ANALYSIS

ENCLOSURE 4