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COOPER NUCLEAR STATION
TECHNICAL EVALUATION REPORT
ON THE INDIVIDUAL PLANT EXAMINATION
BACK-END SUBMITTAL

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ENCLOSURE 3

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E. Executive Summary

SCIENTECH performed a submittal-only review of the back-end portion of the Nebraska Public Power District's Individual Plant Examination (IPE) on the Cooper Nuclear Station (CNS).

E.1 Plant Characterization

CNS consists of a General Electric Company designed BWR-4 with a Mark I containment. CNS operates at thermal power levels up to 2381 MW and provides an electrical output of 801 MW (gross) and 778 MW (net). CNS's operating license was issued on January 18, 1974, and CNS began commercial operation on July 1, 1974.

E.2 Licensee IPE Process

The CNS IPE was performed based on Level 1 and Level 2 probabilistic risk assessments (PRAs) using the techniques set out in NUREG/CR-2300 and NUREG-1150. The IPE team consisted of in-house staff and consultants of Science Application International Corporation (SAIC). Independent, in-house personnel and an independent consultant from the technical services company, ERIN, Inc., reviewed the IPE submittal.

The CNS IPE team developed containment event trees to analyze the containment performance under severe accidents. The IPE team used the MAAP computer program to determine plant damage state behavior (success or failure), accident timing, radionuclide release fractions, and radionuclide decontamination factors.

E.3 Back-End Analysis

For CNS, the IPE team calculated a total core damage frequency of 7.1E-5 per year,[1] which is higher than the 4.5E-6 per year calculated for Peach Bottom (the NUREG-1150 Mark I plant) but is consistent with the frequencies per year calculated for Browns Ferry and Dresden. The conditional probabilities of containment failure, given core damage, were 36 percent for early failures and 31 percent for late failures. The probability of no failure or venting at CNS was calculated to be 33 percent. During the back-end analysis only venting through the 2-inch vent lines was modeled. For all plant damage states, the conditional probability of venting (given core damage) was 0.28. This vent path was not considered a containment failure because it was performed via the standby gas treatment system.

The team found the drywell liner melt-through to be the dominant contributor to the early containment releases resulting from severe accidents. The liner melt-through accounted for 99 percent of early releases.

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[1] In going from the Level 1 PRA analysis to the Level 2 analysis, the CNS IPE team adjusted the core damage frequency from 7.97E-5/year to 7.1E-5/year by eliminating the service water scenario. This correction was made only to the Level 2 analysis because time did not permit a detailed reanalysis of the Level-1 results (note pages 3-782-3 of the submittal).

E.4 Containment Performance Improvements (CPI)

CNS added to its design configuration a hardened vent system, designed so that a torus purge line directs flow to the hard pipe vent system, which contains an air-operated isolation valve and a rupture disk. The discharge from the latter system is directed to the Standby Gas Treatment System discharge and into the Elevated Release Point stack. When the Emergency Operating Procedures so direct, CNS performs containment venting, doing so manually and in accordance with Emergency Support Procedure 5.8.18, "Primary Containment Venting for PCPL." CNS did not evaluate the potential benefits of 1) providing an alternate water supply for the drywell sprays and 2) enhancing the reactor pressure vessel depressurization system. Both of these evaluations are to be performed when the upgrade to the probabilistic safety assessment is performed.

E.5 Vulnerabilities and Plant Improvements

The new CNS Emergency Operating Procedures allow for earlier spraying of the containment, thus providing the water necessary to mitigate some of the liner melt-through accidents. The IPE team did not take credit for these Operating Procedures because they post-dated the IPE freeze date.

The CNS's definition of vulnerability was based on the closure criteria for individual plant examination of internal events, as provided in NUMARC 91-04, "Severe Accident Issue Closure Guidelines," January 1992, and varied based on relative aspects of core damage frequency, containment bypass, and release significance. The identification and resolution of plant vulnerabilities involved assessing the overall safety impact of candidate safety enhancements. Cost-benefit considerations also were addressed before any of the proposed changes were implemented.

E.6 Observations

Based on the review, SCIENTECH noted the following strength in the Cooper IPE back-end analysis:

- The CNS IPE team performed containment analysis using detailed CETs. The main CET consisted of 13 (main) top events which (1) directly influenced the accident progression, or (2) directly influenced the source term. The IPE team developed 13 subtrees to represent the logic used to quantify each of the above CET top events. These trees were made up of basic events, phenomenological events, human actions, pre-existing conditions, and sub-subtrees. Seven sub-subtrees were used to quantify some of the top events in the subtrees. The sub-subtrees were made up of basic events, phenomenological events, human actions, pre-existing conditions, and previously developed subtrees.

1. INTRODUCTION

1.1 Review Process

This technical evaluation report (TER) documents the results of the SCIENTECH review of the back-end portion of the Cooper Nuclear Power Plant Individual Plant Examination (IPE) submittal. [1, 7] This TER was prepared to comply with the requirements for IPE back-end reviews of the U.S. Nuclear Regulatory Commission (NRC) in its contractor task orders, and adopts the NRC review objectives, which include the following:

- To help NRC staff determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335
- To help NRC staff assess if the IPE submittal meets the intent of Generic Letter 88-20
- To complete the IPE Evaluation Data Summary Sheet

A draft TER for the Back-End portion of the Cooper IPE submittal was submitted by SCIENTECH to NRC in November 1993. Based in part on this draft submittal, the NRC staff submitted a Request for Additional Information (RAI) to Nebraska Public Power District on October 21, 1994. Nebraska Public Power District responded to the RAI in a document dated February 20, 1995. This final TER is based on the original submittal and the response to the RAI.

Section 2 of the TER summarizes SCIENTECH's review and briefly describes the Cooper IPE submittal, as it pertains to the work requirements outlined in the contractor task order. Each portion of Section 2 corresponds to a specific work requirement. Section 2 also outlines the insights gained, plant improvements identified, and utility commitments made as a result of the IPE. Section 3 presents SCIENTECH's overall observations and conclusions. References are given in Section 4. The appendix contains an IPE evaluation and data summary sheet.

1.2 Plant Characterization

CNS consists of a General Electric Company designed BWR-4 with a Mark I containment. CNS operates at thermal power levels up to 2381 MW and provides an electrical output of 801 MW (gross) and 778 MW (net). CNS's operating license was issued on January 18, 1974, and went into commercial operation on July 1, 1974.

The primary containment consists of the traditional inverted light bulb steel drywell and steel torus wetwell design with a suppression pool of water typical of the Mark I design. The containment was designed for 60 psig internal pressure, has a total free volume space of about 242,550 cubic feet (132,250 cubic feet in the drywell and 110,300 above the water pool in the Torus), and a nominal 87,650 cubic feet of water in the suppression pool.

The secondary containment consists of the reactor building, which surrounds the primary containment and contains refueling equipment and spent fuel storage facilities. It is a reinforced concrete structure with steel framing enclosed with insulated metal siding, above the refueling floor. The fasteners used to attach the siding to the framing will yield at 0.527 psi, relieving internal building pressure.

2. TECHNICAL REVIEW

2.1 License IPE Process

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

2.1.1 Completeness and Methodology.

The CNS IPE back-end submittal is essentially complete with respect to the level of detail requested in NUREG-1335. The submittal appears to meet the NRC sequence selection screening criteria described in Generic Letter 88-20.

The IPE methodology used is described clearly. The approach followed is consistent with the basic tenets of Generic Letter GL 88-20, Appendix 1.

As noted in Section 1.3.1, page 1-8 of the submittal, the CNS IPE followed the techniques as covered in NUREG/CR-2300 [2] and the NUREG-1150 program. [3] The Level 1 model quantification led to identification of 12 plant damage states (PDSs) of which eight with frequencies greater than 1E-6 per year were selected for Level 2 analysis. As noted in Section 2.2.1 of this report, one PDS was found not to lead to core damage and therefore was dropped from the Level 2 analysis. Containment event trees (CETs) were developed for the remaining seven PDSs. The MAAP computer code was used to calculate severe accident event timing and containment loads for representative sequences.

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

The IPE team collected plant design and operating data applicable to the severe accident progression at CNS. The team reviewed and verified these data, to the extent possible, through a containment walkdown and interviews with plant personnel.

2.1.3 Licensee Participation and Peer Review.

The CNS IPE team consisted of in-house staff and consultants of Science Application International Corporation (SAIC). Independent in-house personnel and an independent consultant from the technical services company, ERIN, Inc., reviewed the IPE submittal. As noted in Section 5.2, page 5-2 of the submittal, the in-house review of the Level 2 study was not as intensive as it was of the Level 1 study. The submittal notes that reviewer comments on the IPE (which we assume focused primarily on the front-end portion of the IPE) and their resolutions were extensive and well documented; the submittal reports none of the comments or comment resolutions.

2.2 Containment Analysis/Characterization

2.2.1 Front-end Back-end Dependencies.

The CNS IPE team coupled the front-end analysis to the back-end analysis by binning the front-end dominant sequences into a few groups of PDSs with similar back-end characteristics. The PDS analysis was performed in two phases. The first phase involved sorting and grouping the cutsets from the dominant accident sequences into 30 preliminary PDSs. The second phase involved consolidating with only small differences the preliminary PDSs into 12 final ones as follows:

- SBO—station blackout accident
- LOCA—loss of coolant accident
- AWHP—anticipated transient without a scram (ATWS) with vessel failure at high pressure
- AWLP—ATWS with vessel failure at low pressure
- TQUX—transient with failure of high-pressure coolant injection (HPCI) and failure to depressurize the vessel for low-pressure coolant injection (LPCI)
- TQUV—transient with loss of all HPCI, successful vessel depressurization, but failure of LPCI
- TPUV—transient with stuck open relief valve (SORV) and failure of both HPCI and LPCI
- TPW—transient with an SORV, initial coolant injection, but failure of RHR
- TPUX—transient with SORVs, failure of HPCI, and failure to depressurize the vessel
- TWDT—transient with the immediate loss of reactor equipment cooling system and failure of the power conversion system (condenser) and residual heat removal (RHR) system; vessel failure occurs between 4 and 24 hours
- TWLT—same as TWDT, except that vessel failure occurs after 24 hours
- SW—transient with loss of service water

Table 3.1.5-3, on pages 3-417 and 3-418, has the results of the final PDS binning of accident sequences. Of the 12 PDSs, four (LOCA, AWHP, TPW, and TWLT) had core damage at frequencies less than 1E-6 per year and therefore the team did not consider them in its back-end analysis.

The PDS "SBO" was divided into the following four PDSs by dropping the "O" and adding the following to "SB": 1) the prefixes "ST" or "DT" to represent short or delayed time and 2) the suffixes "HP" or "LP" to represent high or low primary system pressure at the onset of core damage, (i.e., ST-SBHP, ST-SBLP, DT-SBHP, and DT-SBLP).

As discussed in Section 3.5, pages 3-782 and 3-783 of the submittal, the back-end analysis revealed that an error had occurred in analyzing the sequences of the PDS "SW." Once the error was corrected, the PDS SW did not lead to core damage and therefore was not considered in the back-end analysis. The remaining seven final PDSs were considered.

The CNS IPE team defined the PDS conditions related to the availability of plant equipment and the conditions in the reactor pressure vessel by answering the following 13 questions (Section 4.3.2, pages 4.3-2 and 4.3-3):

- Is there a stuck open safety valve?
- Are the safety relief valves operable?
- Is the reactor pressure vessel depressurized before core damage?
- Does containment pressure cause the relief valves to close?
- Is AC power available?
- Is DC power available?
- Was the containment vented with 24-inch vent lines?
- Are the control rod drive pumps and system operational?
- Are the emergency core cooling systems (ECCSs) operational or available to mitigate post-accident core damage?
- Are alternate injection systems available?
- Are containment venting systems available?
- Is venting being performed through the drywell?
- Is the Standby Gas Treatment System available?

The PDSs determined the accident progression through the answers to the above questions, which were used to set the initial conditions. Answers to these questions for each PDS are listed in Table 4.3.3, pages 4.3-5 and 4.3-6 of the submittal. (This table does not reference two PDSs -- "TQUX" and "TPUV" -- which were found to end with no vessel breaches.)

The CNS IPE team appears to have conveyed the important vessel and plant equipment conditions from the front-end analysis to the back-end analysis.

2.2.2 Sequences with Significant Probabilities.

In the back-end analysis, the CNS IPE team examined sequences with frequencies greater than 1E-6 per year. (See Table 4.3-1, page 4.3-1 of the submittal.) It appears that the CNS IPE met the sequence selection criteria, as outlined in Appendix 2 to Generic Letter 88-20.

2.2.3 Failure Modes and Timing.

Section 4.4 of the submittal characterizes the CNS containment failure. For purposes of comparison, the CNS IPE team used the containment strength analysis of the Peach Bottom containment [5] performed by Chicago Bridge and Iron to assess containment strength and the magnitude of loads necessary to cause the containment to fail. [6] The following containment components were examined at CNS and specifically compared for differences with components at Peach Bottom (page 4.4-1 of the submittal):

- Drywell shell
- Vent line
- Suppression chamber shell
- Vent line bellows
- Drywell head
- Drywell head flange bolts
- Containment hatches
- Mechanical penetrations (piping)
- Electrical penetration assemblies

The CNS IPE team found three failures to have a significant bearing on the plant-specific modeling of containment response to estimated severe accident loads. The first failure was of the head gasket, which leaked in a steam environment when internal pressures were greater than 23 psig at high temperatures (above 500°F). The leak areas for the range of expected pressures are given in Section 4.4.1, pages 4.4-1 and 4.4-2 of the submittal.

The second and third failures were at the drywell head and just above the torus equator. At low drywell temperature, failure above the torus equator occurred at a median pressure of 175 psig. Figure-4.4-1, page 4.4-2, gives the cumulative probability of failure for the drywell head and torus for varying pressures and for temperatures less than 500°F.

The MAAP computer code was used to calculate possible times of failure of the above components under predicted containment loads.

The CNS IPE team found that the dominant contributor to the early containment release was drywell liner melt-through, which accounted for 99% of early failures resulting from severe accidents (Sections 1.4.5.4.2.1 and 1.4.5.4.2.2).

CNS has installed 25 electrical penetration assemblies (EPAs) (20 - General Electric, 4 - Conax, and 1 - ISTC). These EPAs are expected to maintain structural and leak integrity up to at least 700°F, below which temperature containment failure occurs at the drywell head flange.

The CNS IPE team appears to have addressed containment failure modes with regard to all phenomena relevant to the CNS.

2.2.4 Containment Isolation Failure.

The CNS IPE team quantified and assessed containment bypass and isolation failures, as indicated in Section 4.4.2.4, page 4.4-8 of the submittal. [4] Calculations showed that bypass and isolation failures were not significant at CNS and therefore were not included in the CETs (Section 4.4.2.5, page 4.4-9).

2.2.5 System/Human Responses.

The CNS CETs consist of top events to analyze system/human responses that could affect the progression of severe accidents or have a bearing on the radiological consequences of accidental radionuclides releases. The following is a list of such top events, which appears in Table 4.6-2 along with the split fraction values (pages 4.6-15 through 4.6-25 of the submittal):

- AC power is not restored early.
- Late AC power is not restored when the event is a station blackout.
- Alternate flow (core injection with condensate or service water) is not initiated.
- Operator fails to depressurize after core damage.
- Human error results in a failure to restore control rod drive (CRD) system flow to the vessel.
- Operator fails to provide CRD flow to debris.
- Operator fails to actuate sprays early to depressurize containment.

- Operator fails to actuate sprays to cool debris.
- Operator fails to vent containment early.
- Operator fails to vent containment late.

2.2.6 Radionuclide Release Characterization.

The CNS IPE team characterized radioactive releases for seven PDSs: "AWLP," "ST-SBLP," "DT-SBLP," "DT-SBHP," "TWDT," "TQUV," and "TPUX." The IPE team found that, although their occurrence did result in core damage, two PDSs, i.e., "TQUX" and "TPUV," did not challenge the integrity of the reactor vessel or the containment. CETs and source term models were not developed for these two PDSs, but their frequencies per year were added to the final release frequencies.

The end states of the CETs were categorized into 12 release groups, according to their severity and time of release. Another group consisted of CETs whose late releases each amounted to less than 1E-4 percent of radionuclides, excluding noble gases. This group represented the "no containment failure" and "no venting" modes. Table 4.7-1, page 4.7-12 of the submittal, defines the 13 release categories used.

The screening criterion provided in GL 88-20, Appendix 2, "Criteria for Selecting Important Severe Accident Sequences" is as follows:

> Any functional sequence that has a core damage frequency greater than or equal to 1.0E-6 per year and that leads to containment failure which results in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release category of WASH-1400.

For purposes of the IPE submittal, the CNS IPE team considered any release to the outside that resulted in more than 1% of any radionuclide group, excluding noble gases, to be a large release. This criterion is more conservative than that recommended in Generic Letter 88-20. To screen for sequences more likely to be risk-significant, the IPE team examined the source term results and selected the sequences with releases of 1% or more of iodine, cesium, tellurium, or strontium. These selected sequences are displayed in bold type in Tables 4.7-3 through 4.7-9 of the submittal.

2.3 Accident Progression and Containment Performance Analysis

2.3.1 Severe Accident Progression.

Section 4.6.1, pages 4.6-1 through 4.6-12 of the submittal, describes the severe accident progression analysis. The IPE team used the MAAP Version 3.0B computer program to determine plant damage state behavior (success or failure), accident timing, radionuclide release fractions, and radionuclide decontamination factors. The drywell liner melt-through, although not explicitly modeled by MAAP, was addressed in a probabilistic manner using containment event trees. The following phenomena were considered as potential contributors to early containment failure (Section 4.7.3.1, page 4.7-25):

- In-vessel steam explosion

- Ex-vessel steam explosion

- Drywell liner melt-through

- High-pressure melt ejection (HPME) combined with suppression pool heatup and in-vessel hydrogen generation

- Reactor pressure vessel (RPV) blowdown combined with in-vessel hydrogen generation.

2.3.2 Dominant Contributors: Consistency with IPE Insights.

Table 1 in this report shows the results of SCIENTECH's comparison of the dominant contributors to the CNS containment failure with those contributors identified during individual plant examinations performed at the Fitzpatrick, Oyster Creek, Browns Ferry, Duane Arnold, and Dresden plants, and with the NUREG-1150 PRA results obtained at Peach Bottom.

As described in Section 4.7.2.2, page 4.7-11 of the submittal, the CNS IPE team categorized possible radionuclide releases at CNS according to the time at which they might occur after the onset of core damage. These included: (1) an early release group defined as releases occurring within 2 hours of the onset of core damage, (2) a delayed release group defined as releases occurring between 2 and 4 hours after the onset of core damage, and (3) a late release group of releases occurring 4 hours after the onset of core damage. In table 1, both early and delayed releases at CNS are combined and shown under the early release group.

No major differences exist among the various results shown in Table 1. According to the table, the probability of no failure, i.e., containment intact, is higher at CNS than at the other plants listed. This is because, at CNS, the values calculated for "no vessel breach" were included as part of those calculated for "intact." When the same values at the other plants are combined, the values are comparable to those for CNS.

2.3.3 Characterization of Containment Performance.

As described in Section 4.5 of the submittal, the CNS IPE team analyzed containment performance using CETs. The main CET consisted of the following 13 (main) top events

Table 1. Containment failure as a percentage of CDF: CNS results compared with the results of the Fitzpatrick, Oyster Creek, Browns Ferry, Duane Arnold, and Dresden IPEs and with the Peach Bottom NUREG-1150 PRA results.

Containment Failure	Fitzpatrick IPE	Oyster Creek IPE	Browns Ferry IPE	Duane Arnold IPE	Dresden IPE	Peach Bottom/ NUREG-1150	Cooper IPE
CDF (per year)	1.9E-6	3.2E-6	4.8E-5	7.8E-6	1.8E-5	4.5E-6	7.1E-5
Early failure	60	16	46	47	3	56	36
Bypass	na*	7	na	0	0	na	0
Late failure	26	26	26	32	86	16	31
Intact	3	0	3	21	11	18	33
No vessel breach	11	51	25	na	na	10	na

*na - Not available

which (1) directly influenced the accident progression, or (2) directly influenced the source term:

- Reactor vessel depressurized, DP
- Early injection recovered, E_INJ
- Reactor vessel failure, VF
- Early containment failure, CFE
- Early release to suppression pool, EPOOL
- Debris cooled, DCOOL
- Containment failed late, CFL
- Late release to pool, LPOOL
- Late injection to reactor vessel, L_INJ
- Drywell sprays activated, DWSPRAYS
- Fission products retained in containment, FPR
- Reactor building retention, RB
- Containment vented, VENT

The top events related to system/human response, which are listed in Section 2.2.5 of this report, do not appear in the above list because they are the top events of subtrees and sub-subtrees. (The subtrees and the sub-subtrees are described below.)

The IPE team developed 13 subtrees to represent the logic used to quantify each of the above CET top events. These trees were made up of basic events, phenomenological events, human actions, pre-existing conditions, and sub-subtrees.

Seven sub-subtrees were used to quantify some of the top events in the subtrees. The sub-subtrees were made up of basic events, phenomenological events, human actions, pre-existing conditions, and previously developed subtrees.

Section 4.6.3 of the submittal describes the CET basic event quantification, and the split fractions appear in Table 4.6-2, pages 4.6-15 through 4.6-25. As shown in the table, basic events were classified into three groups: Classification 1—CET values taken from studies of other similar or generic plants, Classification 2—CET values based on plant-specific computer code calculations, and Classification 3—CET values based on human reliability analyses.

The CNS IPE team appears to have characterized the containment performance in detail.

2.3.4 Impact on Equipment Behavior.

Section 4.6.2, page 4.6-12 of the submittal, describes the survivability of the vessel injection systems under severe accidents environments. The CNS Injection System consists of three discrete systems: the Control Rod Drive System, the Emergency Core Cooling Systems, and the Alternate Injection Systems, all of which are capable of injection during an accident according to the submittal, although their possible failure was not discounted:

The CNS Nuclear Station has undergone an extensive modification to environmentally qualify (EA) components to continue functioning in harsh environments caused by accidents. Even with this EA rating the probability for failure of the systems inside the secondary containment was not dismissed.

For these equipment modifications, which were made before the initiation of the IPE program, the IPE team took credit in the Level 2 analysis such that, when the soft vent system was used, a probability was assigned to the equipment surviving when the soft vent into the reactor building failed.

2.3.5 Uncertainty and Sensitivity Analyses.

The CNS IPE team performed a total of eight sensitivity studies (SSs). SSs 1, 2, and 3 were performed to quantify the uncertainties of AC power recovery. According to Section 4.7.3.4.3, page 4.7-34 of the submittal, the results showed that:

> AC power recovery is critical to the mitigation of the accidents that are initiated by a loss of power. All mitigation systems are powered by AC power; consequently without AC power the probabilities of vessel failure and of a core-concrete interaction are substantially increased.

The IPE team investigated the effects of HPME and RPV blowdown in SS 6 by assuming that a higher probability of failure existed in the drywell than in the torus. The results showed that the frequency of early releases had not changed appreciably.

Uncertainties associated with drywell liner melt-through, HPME, and RPV blowdown were addressed by performing sensitivity studies. According to Section 4.7.3.4.7, page 4.7-57 of the submittal, the results of SS 7 showed that:

> The removal of the [drywell liner] melt-through probability has the effect of deleting the most dominant contributor to early containment failure and early releases. The downside to this is the increase in the core-concrete interaction scenarios which lead to larger releases late in the sequences.

According to Section 4.7.3.4.8, page 4.7-63 of the submittal, the results of SS 8 showed that:

> Containment venting, while resulting in no favorable increase in containment integrity probability, only slightly affects the source term and results in no appreciable increase in the radionuclides released.

2.4 Reducing Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability.

In Section 1.4.5.5, page 1-17 of the submittal, the IPE team noted that:

> There are no significant vulnerabilities associated with the CNS Nuclear Station . . .

The CNS's definition of vulnerability was based on the closure criteria for individual plant examination of internal events as provided in NUMARC 91-04, "Severe Accident Issue Closure Guidelines," January 1992, and varied based on relative aspects of core damage frequency, containment bypass, and release significance. The identification and resolution of plant vulnerabilities involved assessing the overall safety impact of candidate safety enhancements. Cost-benefit considerations also were addressed before any of the proposed changes were implemented.

2.4.2 Plant Improvements.

Section 1.4.5.5, page 1-17 of the submittal, notes the following:

> The study shows that without copious amounts of water on the drywell floor to quench the corium (post vessel breach), the probability of a drywell liner melt-through is greatly increased. The new Emergency Operating Procedures allow for the earlier spraying of the containment, thus providing the water necessary to mitigate some of these accidents.

With the exception of the loss of service water accidents, which were not considered during the back-end analysis (see Section 2.2.1 of this report), the new EOPs were not credited in the IPE. Because Revision 4 of the Emergency Procedure Guidelines, post-dates the design freeze date of the IPE, the version of the EOPs based on Revision 3 of the EPGs was used for the IPE.

2.5 Responses to CPI Program Recommendations

Generic Letter No. 88-20, Supplement No. 1, reiterates the following recommendations made by the Containment Performance Improvement (CPI) Program 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.

Supplement No. 1 also notes that the above improvements should be considered in addition to improvements that stem from the evaluation and implementation of hardened vents.

CNS added to its design configuration a hardened vent system, designed so that a torus purge line directs flow to the hard pipe vent system, which contains an air-operated isolation valve and a rupture disk. The discharge from the latter system is directed to the Standby Gas Treatment System discharge and into the Elevated Release Point stack. When directed per the Emergency Operating Procedures, containment venting is performed manually, using Emergency Support Procedure 5.8.18, "Primary Containment Venting for PCPL." CNS did not evaluate the potential benefits of 1) providing an alternate water supply for the drywell sprays and 2) enhancing the reactor pressure vessel depressurization system. Both of these evaluations are to take place when the upgrade to the probabilistic safety assessment is performed.

2.6 IPE Insights, Improvements, and Commitments

As noted in Section 1.4.5.5, page 1-17 of the submittal, the CNS IPE team calculated a 38 % probability of a large or significant release of radionuclides to the environment, given a core-damage event. The team gained a number of insights into how the consequences of a core damage event at CNS could be mitigated through the presence of the following capabilities, conditions, or practices:

- Capability of the reactor building to retain some of the radionuclides released from the primary containment. (Section 1.4.5.5, page 1-17)

- Early failure of the drywell head seal, which would result in the retention of a significant fraction of radionuclides in the containment, and the use of containment vents to reduce containment pressure and prevent large failures of the containment, thus reducing the amount of radionuclides released. (Section 1.4.5.5, page 1-17)

- Presence of a copious amount of water on the drywell floor to quench corium after vessel breach, which would greatly reduce the probability of drywell liner melt-through. (Section 1.4.5.5, page 1-17)

- Capability of the drywell sprays to effectively reduce pressure and temperature once the sprays were recovered during an accident.

The IPE submittal notes the following plant improvements:

- The CNS has performed extensive modifications to environmentally qualify components to continue functioning in harsh environments under accident conditions. (Section 4.6.2, page 4.6-12)

- The new Emergency Operating Procedures permit earlier spraying of the containment, thus providing the water necessary to mitigate some of the drywell liner melt-through accidents. (Section 1.4.5.5, page 1-17)

- CNS added a hardened vent system to its design configuration.

According to the submittal, the CNS has made no commitments to carry out other plant or procedural changes as a result of the back-end analysis.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

As discussed in Section 2 of this TER, the CNS IPE submittal contains a large amount of back-end information, which contributes to the resolution of severe accident vulnerability issues at CNS. The submittal is well presented and describes what drives the IPE results. The CNS IPE submittal appears to demonstrate an understanding of the impact of severe accidents on containment failures and subsequent radionuclide releases.

The key points of the SCIENTECH submittal-only technical evaluation of the CNS back-end submittal are summarized as follows:

- The submittal appears to be complete in accordance with the level of detail requested in NUREG-1335 and appears to meet the NRC sequence selection screening criterion described in Generic Letter 88-20.

- The IPE methodology used is described clearly. The approach followed is consistent with the basic tenets of Generic Letter GL 88-20, Appendix 1.

- Recognition of a significant early-release containment failure mode, namely drywell liner melt-through, and consideration of procedural modifications to mitigate the consequences, demonstrate an understanding of the IPE program process and purpose.

4. REFERENCES

1. Nebraska Public Power District, "Cooper Nuclear Station Individual Plant Examination Report," March 1993.

2. NUREG/CR-2300, "PRA Procedure Guidelines," ANS/IEEE, January 1983.

3. NUREG-1150, "Severe Accidents Risks: An Assessment for Five U.S. Nuclear Power Plants," June 1989.

4. NEDC 92-053, "Containment Isolation and Bypass Assessment for CNS Level 2 PRA," Rev. 0.

5. CBI NA-CON, Inc., "Mark I Containment Severe Accident Analysis," April 1987.

6. NEDC 92-018, "Comparison of CBI Containment Severe Accident Analysis to CNS Containment," Rev. 0.

7. Nebraska Public Power District, "Response to Request for Additional Information on the Cooper Nuclear Station, Individual Plant Examination Submittal," February 20, 1995.

Appendix
IPE Evaluation and data Summary Sheet

BWR Back-end Facts

Plant Name

Cooper

Containment Type

Mark I

Unique Containment Features

None found

Unique Vessel Features

None found

Number of Plant Damage States

12

Ultimate Containment Failure Pressure

175 psig

Additional Radionuclide Transport And Retention Structures

Reactor building retention credited

Conditional Probability That The Containment Is Not Isolated

Given as negligible. However, an estimated value could not be located in the submittal.

Important Insights, Including Unique Safety Features

The reactor building retains radionuclides. The early failure of the drywell head seal could cause less catastrophic containment failure. The containment venting reduces containment pressure, which prevents large failures.

Appendix (continued)
IPE Evaluation and Data Summary Sheet

Implemented Plant Improvements

The CNS upgraded equipment to qualify it for operation in harsh environments under accident conditions.

The CNS upgraded Emergency Operating Procedures to permit earlier spraying of the containment, thus providing the water necessary to mitigate some of the drywell liner melt-through accidents.

CNS added a hardened vent system to its design configuration.

C-Matrix

PDS	Frequency per year	Early Failure	Late Failure	Intact
AWLP	3.28E-06	0.66	0.34	0
ST-SBLP	1.13-05	0.36	0.36	0.28
DT-SBLP	1.30E-06	0.78	0.15	0.07
DT-SBHP	1.50E-05	0.74	0.25	0.01
TWDT	5.94E-06	0.67	0.33	0
TQUV	3.04E-06	0.67	0.33	0
TPUX	1.89E-05	0	0.45	0.54
TPUV	4.25E-06	0	0	1
TQUX	4.17E-06	0	0	1