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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INDIVIDUAL PLANT EXAMINATION FOR INTERNAL EVENTS, GENERIC LETTER 88-20

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letters dated December 23, 1992, and February 3, 1993, Entergy Operations, Inc., (EOI) submitted the Individual Plant Examination (IPE) for Grand Gulf Nuclear Station Unit 1 (GGNS). The GGNS IPE was in response to Generic Letter (GL) 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities," dated November 23, 1988, and its associated supplements: Supplement 1, August 29, 1989; Supplement 2, March 30, 1990; Supplement 3 (Containment Performance Improvement program), June 29, 1990; and Supplement 4 (External Events), June 27, 1991. The Nuclear Regulatory Commission (NRC) staff employed Sandia National Laboratory (SNL) to review the IPE submittal.

The GGNS plant is a 3833 Megawatt (thermal) General Electric boiling water reactor (BWR) 6 reactor with a Mark III containment.

Guidance to licensees on reporting the results of an IPE for a nuclear power plant was issued by the staff in NUREG-1335, "Individual Plant Examination: Submittal Guidance," dated August 1989.

Because GGNS had already been analyzed by NRC in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," dated June 1989, the staff modified its review of the GGNS IPE results for their reasonableness. For GGNS, SNL's review was based on a comparison between the results for GGNS of the IPE submittal and the NUREG-1150 study documented in NUREG/CR-4550 and NUREG/CR-4551, Level 1 and Level 2 analyses, respectively. The results for GGNS are in two parts of Volume 6 of both NUREG/CR-4550 ("Analysis of Core Damage Frequency from Internal Events: Grand Gulf, Unit 1," dated April 1987) and NUREG/CR-4551 ("Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," dated December 1991). Plants that had not received such an analysis and review were sent requests for additional information.

The review involved the efforts of SNL and focused on whether EOI's IPE method was capable of identifying severe accident vulnerabilities for GGNS in the IPE. Therefore, SNL's review considered (1) the completeness of the information in the IPE and (2) the reasonableness of the results given the GGNS design and operation. A summary of SNL's findings and the staff's

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conclusions are provided below. The details of SNL's findings are in the technical evaluation report (TER). The TER contains the core damage estimation (front end analysis), the core performance (back-end analysis), and the human reliability analysis for GGNS.

In accordance with GL 88-20, EOI proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal [DHR] Requirements." No other specific USIs or generic safety issues were proposed for resolution as part of the GGNS IPE. A summary of the staff's conclusions are provided below.

The acronyms not defined in this evaluation and the reference to NUREG/CR-4550 are listed in the TER.

II. EVALUATION

Based on the TER, the staff found that the IPE, NUREG/CR-4550, and NUREG/CR-4551 analyses used the same techniques; however, the IPE reflects more current plant configuration and procedures. This new information led to analyses of several support system initiators that were not quantified in the NUREG/CR-4550 analysis. The primary differences between the IPE and the NUREG-1150 study for GGNS are the following:

- The NUREG/CR-4550 analysis screened out all special initiators except loss of instrument air.
- The dependency of standby service water (SSW) upon SSW pump house ventilation was not modeled in NUREG/CR-4550 because each pump house is away from other buildings (near the cooling tower) and normally has open louvers on the walls. This configuration, along with the air current from the proximity of the cooling tower, was assumed to provide ample ventilation. In the IPE, the louvers are normally closed.
- The plant-specific failure rate for the reactor core isolation cooling (RCIC) turbine-driven pump in the IPE submittal is significantly higher than the generic value used in NUREG/CR-4550.
- The dominant mechanism of early containment failure in the IPE, venting of the reactor pressure vessel (RPV) through the main steam isolation valves (MSIVs) (which accounts for 38 percent of the conditional containment failure probability), was not considered in the NUREG/CR-4551 analysis.
- The NUREG/CR-4550 analysis did not include an internal flooding analysis.

On the basis of these findings, the staff concludes that the differences in the results of the IPE and NUREG-1150 study are reasonable. In NUREG/CR-4550, a mean core damage frequency (CDF) of 4E-6/reactor year was estimated for GGNS and the CDF was dominated by station blackout (94%). EOI has estimated an CDF of 2E-5/reactor-year from internally initiated events in the IPE, including a contribution from internal floods of 2E-7/reactor-year. Loss of offsite power

(LOSP) contributes 55%, loss of Division 2 ac bus 19%, loss of Division 1 ac bus 6%, transient with loss of power conversion system (PCS) 4%, transient with PCS initially available 3%, loss of Division 2 dc bus 2%, and loss of Division 1 dc bus 2%.

The important system and equipment contributions to the estimated CDF that appear in the top sequences are the common-cause failure (CCF) of the following:

- batteries A, B or C
- SSW motor-operated discharge valve F005B
- diesel generator (DG) room ventilation components
- SSW discharge check valves
- automatic depressurization system (ADS) and non-ADS safety valves
- SSW pump house dampers

failure to depressurize in the short term, and load shedding and sequencing system inadvertent load shed (Division 2).

To analyze internal flooding, EOI performed walkdowns, identified flood zones, and developed and screened flood scenarios. The flood scenarios were based on flood sources, flood propagation, and flood impact on mitigating and support systems. Quantification of these scenarios resulted in a flooding related CDF of 2E-7/reactor year. Based on the enclosed TER, the staff found EOI's flooding analysis in the IPE reasonable.

Regarding the resolution of USI A-45, EOI stated in the IPE submittal that by comparing the total CDF of 2E-5/reactor year for GGNS with quantitative criteria used by NRC, it concluded that GGNS has no unique DHR vulnerabilities. USI A-45 is discussed in NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 19, dated June 1995. It was initiated to evaluate the safety adequacy of the decay heat removal function in operating light water reactors (LWRs) and the staff selected a goal, or quantitative criteria, that the CDF due to failure of the DHR function would be less than 1E-5/reactor year. Based on the discussions of the CDF, including the contributions to the CDF, in the enclosed TER and that the total CDF for GGNS reported in the IPE is 2E-5/reactor year, the staff concludes that USI A-45 has been resolved for GGNS.

EOI also performed a human reliability analysis (HRA) to document and quantify potential failures in human system interactions and to quantify human-initiated recovery of failure events. It identified the following major operator actions as important to prevent core damage or containment failure:

- Inject with firewater
- Bypass high steam tunnel temperature isolation
- Recover offsite power in 10 hours
- Recover diesel hardware failures within 1 hour
- Depressurize the reactor vessel
- Recover offsite power in 4 hours
- Restore SSW train-A pump room ventilation

- Recover dc hardware failure
- Recover diesel from maintenance in 1 hour
- Properly restore high-pressure core spray (HPCS) system
- Vent the RPV through the MSIVs (which results in an intentional bypass of the containment)
- Vent containment
- Activate the hydrogen ignition system.

EOI evaluated and quantified the results of the severe progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. Its back-end analysis appeared to have considered important accident phenomena. The probabilities of containment failure (assuming core damage) are similar in the IPE and the NUREG-1150 study. However, different mechanisms are responsible for early containment failure.

In the IPE submittal, the dominant mechanism for early containment failure is the venting of the RPV through the MSIVs (accounts for 38 percent of the conditional containment failure probability). This mechanism was not considered in the NUREG/CR-4551 analysis. Other conditional containment failure mechanisms (and their associated probabilities) in the IPE are late containment failure (26%), early containment failure (7%), and containment venting (6%). In contrast, in the NUREG/CR-4551 analysis, the dominant mechanism of early containment failure was combustion of hydrogen. EOI's response to the Containment Performance Improvement (CPI) Program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3. This is discussed in Section 2.5.3 of the TER.

Some insights and unique plant safety features identified at GGNS are as follows:

- Cross-tie capability of SSW-B system to the B-injection line of the low-pressure coolant injection.

- Makeup capability to the RPV using the firewater system.

- Reduced potential for CCF because of the difference in Division 3 HPCS DG design and size than those of Divisions 1 and 2.

- Cross-tie capability of the HPCS DG to either Division 1 or 2 during station blackout events per off normal event procedure.

- Improved plant ability to cope with internal floods due to the highly compartmentalized nature of GGNS' auxiliary building.

EOI defined a vulnerability as a core damage sequence with a CDF of 1E-4/reactor-year or greater than 50%, and a containment bypass sequence of 1E-5/reactor-year or greater than 20%. On the basis of this definition, EOI did not identify any vulnerabilities for GGNS. This definition of vulnerabilities is acceptable.

Plant improvements at GGNS were identified for implementation or for further consideration. The following improvements were identified for implementation:

1. Revise the offsite power off-normal event procedure to allow bypass of Level 2 signal in order to allow Division 3 power cross-tie with Divisions 1 and 2.

2. Improve the procedure for bypassing the RCIC leak detection system trip when PSW is unavailable and no steam leak has occurred.

The following improvements were identified for further consideration:

1. Change containment isolation requirements to allow bypassing the containment isolation signal upon loss of Division 1 or 2 ac or dc bus and the re-opening of the containment isolation valves in order to retain the availability of plant service water and instrument air.

2. Improve operator training on control room indication changes to SSW pump house ventilation status.

3. Improve operator training on alternate operation of low-pressure emergency core cooling system pumps to minimize impact of SSW dependence.

4. Modify the portion of the emergency operating procedures that directs the operators to vent the reactor vessel even if core damage has occurred.

III. CONCLUSION

Based on the TER, the staff notes that (1) the GGNS IPE is complete with regard to the information requested by GL 88-20 (and associated guidance in NUREG-1335) and (2) the IPE results are reasonable given the design and operation of GGNS. As a result, the staff concludes that EOI's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and, therefore, that the GGNS IPE has met the intent of GL 88-20.

It should be noted that the staff's review primarily focused on EOI's ability to examine GGNS for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of EOI's detailed findings (or quantification estimates) that stemmed from the IPE. Therefore, this safety evaluation does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

Principal Contributor: E. Lois

Date: March 7, 1996

I. INTRODUCTION

On December 23, 1992 Entergy Operations, Inc., submitted the Individual Plant Examination (IPE) for Grand Gulf Nuclear Station Unit 1 in response to Generic Letter (GL) 88-20 and associated supplements. Because Grand Gulf Unit 1 has already been analyzed by NRC in the NUREG-1150 study, the staff modified the "Step 1" IPE submittal review procedure. Consequently, for Grand Gulf, the contractor's review is based on a comparison between the results of the IPE submittal and the results of the NUREG-1150 study (which is documented in NUREG/CR-4550 and NUREG/CR-4551, Level 1 and Level 2 analyses, respectively). Unlike plants that have not received such analysis and review, requests for additional information were not sent to the licensee during the review.

The modified Step 1 review involved the efforts of Sandia National Laboratories (SNL) and focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the contractor's review considered (1) the completeness of the information and (2) the reasonableness of the results given the Grand Gulf Unit 1 design and operation. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. A summary of contractor's findings and the staff's conclusions are provided below. Details of the contractor's findings are in the attached technical evaluation report (TER) (appendix).

In accordance with GL 88-20, Entergy proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal [DHR] Requirements." No other specific USIs or generic safety issues were proposed for resolution as part of the Grand Gulf IPE.

II. EVALUATION

The Grand Gulf Nuclear Station Unit 1 is a General Electric BWR 6 reactor with a Mark III containment. The staff found that the IPE and the NUREG/CR-4550 analyses used the same techniques in their analyses. The IPE, however, reflects more current plant configuration and procedures. This new information led to analysis of several support system initiators that were not quantified in the NUREG/CR-4550 analysis. The primary differences between the IPE and the NUREG-1150-related study are:

- The NUREG/CR-4550 analysis screened out all special initiators except loss of instrument air.
- The dependency of SSW upon SSW pump house ventilation was not modeled in NUREG/CR-4550 because each pump house is away from other buildings (near the cooling tower) and normally has open louvers on the walls. This configuration, along with the air current from the proximity of the cooling tower, was assumed to provide ample ventilation. In the IPE, more current information indicated the louvers on the walls are normally closed.
- The plant-specific failure rate for the reactor core isolation coolant (RCIC) turbine-driven pump in the IPE submittal is significantly higher than the generic value used in NUREG/CR-4550.

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- The dominant mechanism of early containment failure in the IPE, venting of the reactor pressure vessel (RPV) through the main steam isolation valves (MSIVs) (which accounts for 38 percent of the conditional containment failure probability), was not considered in the NUREG/CR-4551 study.

- The NUREG/CR-4550 analysis did not include an internal flooding analysis.

On the basis of these findings the staff concludes that the differences in the results of the two studies are reasonable. In NUREG/CR-4550, a mean CDF of 4E-6/reactor year was estimated that was dominated by station blackout (94%). The IPE has estimated a core damage frequency (CDF) of 2E-5/reactor-year from internally initiated events, including a contribution from internal floods of 2E-7/reactor-year. Loss of offsite power (LOSP) contributes 55%, loss of Division 2 ac bus 19%, loss of Division 1 ac bus 6%, transient with loss of power conversion system (PCS) 4%, transient with PCS initially available 3%, loss of Division 2 dc bus 2%, and loss of Division 1 dc bus 2%.

The important system/equipment contributions to the estimated CDF that appear in the top sequences are common-cause failure (CCF) of batteries A, B or C; CCF of standby service water (SSW) motor-operated discharge valve F005B; CCF of diesel generator (DG) room ventilation components; CCF of SSW discharge check valves; failure to depressurize in the short term; CCF of the automatic depressurization system (ADS) and non-ADS safety valves; CCF of SSW pump house dampers; and load shedding and sequencing system inadvertent load shed (Division 2).

To analyze internal flooding, the licensee performed walkdowns; identified flood zones; and developed and screened flood scenarios based on flood sources, flood propagation, and flood impact on mitigating and support systems. Quantification of these two scenarios resulted in a flooding related CDF of 2.0E-7/reactor year. The staff found the licensee's flooding analysis reasonable.

Regarding resolution of USI A-45, the IPE submittal states that by comparing Grand Gulf's total CDF of 2E-5/reactor year with quantitative criteria used by the NRC, it is concluded that Grand Gulf has no unique DHR vulnerabilities. On the basis of the staff's review of the licensee's IPE process and Grand Gulf's plant-specific features, the staff finds the licensee's DHR evaluation to be an acceptable resolution of USI A-45.

The licensee performed a human reliability analysis (HRA) to document and quantify potential failures in human system interactions and to quantify human-initiated recovery of failure events. The licensee identified the following operator actions as important: inject with firewater, bypass high-steam tunnel temperature isolation, recover offsite power in 10 hours, recover diesel hardware failures within 1 hour, depressurize the reactor vessel, recover offsite power in 4 hours, restore SSW train-A pump room ventilation, recover dc hardware, recover diesel from maintenance in 1 hour, properly restore the high-pressure core spray (HPCS) system, vent the RPV through the

MSIVs (which results in an intentional bypass of the containment), vent containment, and activate the hydrogen ignition system.

The licensee evaluated and quantified the results of the severe progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. The licensee's back-end analysis appeared to have considered important accident phenomena. The probabilities of containment failure (assuming core damage) are similar in the two studies. However, different mechanisms are responsible for early containment failure. In the IPE submittal, the dominant mechanism of early containment failure is the venting of the RPV through the MSIVs (accounts for 38 percent of the conditional containment failure probability). This mechanism was not considered in the NUREG/CR-4551 study. Other conditional containment failure mechanisms (and their associated probabilities) in the IPE are late containment failure (26%), early containment failure (7%), and containment venting (6%). In contrast, in the NUREG/CR-4551 study, the dominant mechanism of early containment failure was combustion of hydrogen. The licensee's response to Containment Performance Improvement (CPI) Program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3.

Some insights and unique plant safety features identified at Grand Gulf Unit 1 are as follows:

1. Cross-tie capability of SSW-B system to the B-injection line of the low-pressure coolant injection

2. Makeup capability to the RPV using the firewater system

3. Reduced potential for CCF because of the difference in Division 3 HPCS DG design and size than those of Divisions 1 and 2

4. Cross-tie capability of the HPCS DG to either Division 1 or 2 during station blackout events per off normal event procedure

5. Improved plant ability to cope with internal floods due to the highly compartmentalized nature of Grand Gulf's auxiliary building.

The licensee defined a vulnerability as a core damage sequence with a CDF of 1E-4/reactor-year or greater than 50%, and a containment bypass sequence of 1E-5/reactor-year or greater than 20%. On the basis of this definition, the licensee did not identify any vulnerabilities. Plant improvements, however, were identified for implementation or for further consideration. The following improvements were identified for implementation:

1. Revise the offsite power off-normal event procedure to allow bypass of Level 2 signal in order to allow Division 3 power cross-tie with Divisions 1 and 2.

2. Improve the procedure for bypassing the RCIC leak detection system trip when PSW is unavailable and no steam leak has occurred.

The following improvements were identified for further consideration:

1. Change containment isolation requirements to allow bypassing the containment isolation signal upon loss of Division 1 or 2 ac or dc bus and the re-opening of the containment isolation valves in order to retain the availability plant service water and instrument air.

2. Improve operator training on control room indication changes to SSW pump house ventilation status.

3. Improve operator training on alternate operation of low-pressure emergency core cooling system pumps to minimize impact of SSW dependence.

4. Modify the portion of the emergency operating procedures that directs the operators to vent the reactor vessel even if core damage has occurred.

III. CONCLUSION

On the basis of these findings, the staff notes that (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance NUREG-1335) and (2) the IPE results are reasonable given the design, operation, and history of Grand Gulf Unit 1. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and therefore, that the Grand Gulf Unit 1 IPE has met the intent of GL 88-20.

It should be noted that the staff's review primarily focused on the licensee's ability to examine Grand Gulf Unit 1 for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

APPENDIX

GRAND GULF NUCLEAR STATION UNIT 1

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT