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April 23, 1996

U.S. Nuclear Regulatory Commission
Mail Station P1-37
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
1995 Grand Gulf Nuclear Station (GGNS) Annual Environmental Operating Report (AEOR)

GNRO-96/00046

Gentlemen:

Attached is the Grand Gulf Nuclear Station (GGNS) Annual Environmental Operating Report (AEOR) for the period January 1, 1995 through December 31, 1995. This report is submitted in accordance with the Environmental Protection Plan, Appendix B to the GGNS Operating License (NPF-29), Section 5.4, "Station Reporting Requirements".

If you have any questions or require additional information concerning this report, please contact Michael J. Larson at (601) 437-6685, or this office.

Yours truly,

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GRAND GULF NUCLEAR STATION

1995 ANNUAL ENVIRONMENTAL OPERATING REPORT

PREFACE

The Annual Environmental Operating Report (AEOR) provides information and data obtained from implementation of Grand Gulf Nuclear Station's (GGNS) Environmental Protection Plan (EPP), Appendix B to the GGNS Operating License (NPF-29), which only requires terrestrial issues to be addressed, for the period January 1 through December 31, 1995.

The GGNS Final Environment Statement did not identify any aquatic issues. Consequently, the EPP does not address any. The GGNS National Pollutant Discharge Elimination System (NPDES) Permit issued by the Mississippi Department of Environmental Quality (MDEQ) contains effluent limitations and monitoring requirements for aquatic matters. The MDEQ regulates matters involving water quality and aquatic biota.

This report addresses only those issues required by the EPP. In the past, the AEOR included activities associated with the GGNS Construction Permit, and an Updated Final Safety Analysis Report (UFSAR) requirement which involved reporting regional and perched groundwater levels and precipitation data in the AEOR. However, the Nuclear Regulatory Commission approved cancellation of Construction Permit CPPR-119 for Unit 2 on August 21, 1991 (GNRI-91/00176), and GGNS deleted the UFSAR AEOR reporting requirement in 1993 (GNRI-93/00025); therefore, GGNS terminated reporting activities associated with these items.

TABLE OF CONTENTS

			PAGE
PREFACE			ii
SECTION	TOPIC		
1.0	INTRODUCTION		1
	1.1	Impact Assessment and Summary	1
2.0	ENVIRONMENTAL SURVEILLANCE ACTIVITIES		1
	2.1	Transmission Line Surveys	1
	2.2	Cooling Tower Drift Program	1
	2.3	Environmental Evaluations	1
3.0	OBSERVATIONS AND DISCUSSIONS		2
	3.1	Environmental Evaluations	2
4.0	ADMINISTRATIVE REQUIREMENTS		2
	4.1	EPP Changes	2
	4.2	EPP Noncompliances	2
	4.3	Nonroutine Reports	2
	4.4	Potentially Significant Unreviewed Environmental Issues	2

1.0 INTRODUCTION

1.1 Impact Assessment and Summary

GGNS personnel monitored the environmental impact of plant operational activities between January 1 and December 31, 1995. The monitoring results contained in the following sections indicate no adverse impact on the environment due to operation of GGNS. In addition, GGNS personnel have not observed harmful effects or evidence of trends toward irreversible damage to the surrounding environment at GGNS.

2.0 ENVIRONMENTAL SURVEILLANCE ACTIVITIES

2.1 Transmission Line Surveys

GGNS discontinued this program in 1988.

2.2 Cooling Tower Drift Program

GGNS discontinued this program in 1992.

2.3 Environmental Evaluations

The EPP permits changes in GGNS design or operation and performance of tests or experiments that affect the environment, provided they do not involve a change in the EPP or an unreviewed environmental question. However, EPP requirements do not apply to changes, tests or experiments which do not affect the environment. Also, EPP requirements do not relieve GGNS of 10 CFR 50.59 requirements, "Changes, Tests and Experiments," which address the question of safety associated with proposed changes, tests and experiments.

The EPP excludes changes, tests or experiments from the evaluation:

- If all measurable environmental effects confined to onsite areas previously disturbed during site preparation and plant construction, or
- If required to achieve compliance with other federal, state or local requirements.

3.0 OBSERVATIONS AND DISCUSSIONS

3.1 Environmental Evaluations

Review of 1995 environmental evaluations indicate that none of the changes made involved an unreviewed environmental question per the EPP. A review of evaluations conducted did not reveal any potentially significant unreviewed environmental issues. Table 4-1 provides a summary of evaluated changes which could have affected the environment. The evaluations are attached.

4.0 ADMINISTRATIVE REQUIREMENTS

4.1 EPP Changes

GGNS made no changes to the EPP in 1995.

4.2 EPP Noncompliances

GGNS activities contained no EPP noncompliances during 1995.

4.3 Nonroutine Reports

GGNS submitted no nonroutine reports in 1995.

4.4 Potentially Significant Unreviewed Environmental Issues

Review of 1995 environmental evaluations indicated that none of the changes made involved any unreviewed environmental questions per the EPP. A review of evaluations conducted did not reveal any potentially significant unreviewed environmental issues. Table 4-1 provides a summary of evaluated changes which could have affected the environment. The evaluations are attached.

TABLE 4-1

1995 ENVIRONMENTAL EVALUATION SUMMARY

SAFETY AND ENVIRONMENTAL EVALUATION NUMBER	DESCRIPTION
95-0054-R00	The activity involves changing the wording of UFSAR Section 18.1.34 so that minor leakage around vent and drain valves does not have to be eliminated. Instead, such leakage must be maintained as low as practical. Leakage will receive treatment in the normal manner before leaving the plant. Therefore, all releases will continue to be in compliance with established criteria and specifications for the plant. As a result, no unreviewed environmental question exists and a change to the EPP is not required.
95-0057-R00	This activity allowed performing the reactor vessel in-service leak test with the disc removed from 1E12-F050B. The conditions and flow path for the test will remain basically the same. This activity will not result in a release to the environment. As a result there is no unreviewed environmental question and no need to change the Environmental Protection Plan (EPP).
95-0059-R00	This activity revises the maximum allowable stroke time for various primary and secondary containment isolation valves. Release from the plant must continue to comply with established criteria and specifications. This activity is not increasing the probability, quantity or consequences of a steam release beyond that previously evaluated in the FES. As a result, no unreviewed environmental question exists and no change to the EPP is required.
95-0060-R00	This change removes the commitment to submit a summary startup report. Since there will be no release to the environment or change in power level, there is no unreviewed environmental question and no change to the EPP is required.
95-0075-R00	This change deletes the requirement to perform Type C local leak rate testing on nine test connection valves. Functional operation of equipment will not be altered by deleting this testing requirement. As a result, there will be no change in effluents or power level. Therefore the activity will not involve an unreviewed environmental question or require a change to the EPP.
95-0072-R00	This activity allows control rod drive system drive water pressure to be temporarily increased up to 475 psi above reactor pressure during withdrawal. The activity will not change operational design or monitoring and release of effluents so there can be no change in effluents of power levels. As a result, there is no unreviewed environmental question or need to change the EPP.
95-0078-R01	This activity deletes requirement to have a pre-planned alternate method of monitoring when the AXM noble gas radiation monitor is inoperable. Routine monitoring of plant effluents will still occur via established monitoring points and equipment. No effluent limitation is being changed or removed thus there will be no change in effluents or power level and no need to change the EPP.

1995 ENVIRONMENTAL EVALUATIONS

95-066-P&SE.

GRAND GULF NUCLEAR STATION UNIT 1

CHANGES, TESTS OR EXPERIMENTS SAFETY AND ENVIRONMENTAL EVALUATION FORM

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: Scott Kirby DEPT/SECT: P&SE EVAL. #: 95-0054-R00

DOCUMENT EVALUATED: Licensing Document Change Request 95-036

REFERENCES: UFSAR 18.1.34, Technical Specification 5.5 , NUREG -0737

FSAR CHANGE REQUIRED? ☒ Yes ☐ No CR # 95-036

FSAR SECTIONS TO BE REVISED: FSAR Section 18.1.34

TRM CHANGE REQUIRED? ☐ Yes ☒ No

TECH. SPEC. CHANGE REQUIRED? ☐ Yes ☒ No CR # (n/a)

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? ☐ Yes ☒ No

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? ☐ Yes

THE RESPONSIBLE ORGANIZATIONS MUST BE NOTIFIED PRIOR TO IMPLEMENTING THIS CHANGE.

B. Executive Summary (ALSO SERVES AS INPUT TO NRC SUMMARY REPORT)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: UFSAR Section 18.1.34 lists requirements for the integrity of systems outside Containment likely to contain radioactive material for Pressurized Water Reactors and Boiling Water Reactors. The paragraph in this section dealing with Water Leakage requires observable leakage past vent and drain valves be eliminated. The paragrapn dealing with Gas Leakage requires any detected leakage be eliminated. In each instance, the word 'eliminated' will be replaced with 'reduced to as-low-as-practical levels'.

REASON FOR CHANGE, TEST OR EXPERIMENT: Technical Specifications and the UFSAR establish release limits for water and gas leakage to the environment. The requirement to totally eliminate specific leakage paths can cause significant system intrusions with little appreciable affect to total leakage rates.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: This change corrects an overly-burdensome requirement, and is not contrary to the letter or spirit of Technical Specifications or applicable NUREGs. Other testing already in place determines collective leakage for water and gas systems, with corrective action required for leakage above an acceptable amount. No unresolved safety questions are introduced, and this change would not affect the impact of postulated accidents.

II. Safety Evaluation

☐ Not applicable per Safety Evaluation Applicability Review

A. Technical Specifications

1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications.

☐ Yes
☒ No

BASIS: The Technical Specifications, section 5.5, requires controls and procedures to regulate the release of radioactive materials. This UFSAR change does not alter any Tech Spec requirements. Tech Spec section 5.5.2 requires leakage to be minimized to levels as low as practicable, which standard is now being used in this change.

B. Unreviewed Safety Question

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

1. May increase the probability of occurrence of an accident previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: The presence of minimal water and gas leakage is adequately included in current accident evaluations. These leakage pathways pose no new collective impact to accident occurrence probability, and would be tested collectively in other existing testing programs.

2. May increase the consequences of an accident previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: The recycling, treatment, or release of water and gas leakage is covered by current accident evaluations. The addition of leakage past vent and drain valves, or leakage from gas systems, remains within postulated total leakage amounts and changes no design impact. No accident consequences are impacted by the possibility of these leakage pathways.

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: Leakage from vent and drain valves would be collected by liquid radwaste drains, to be treated with all other plant liquid waste. Gas leakage from gas systems and associated piping is treated with other gas leakage. Minimal leakage in these areas will not adversely impact treatment system design capabilities, nor will it impact the probability of an equipment malfunction.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: The consequences of a malfunction of equipment important to safety remain the same as previously evaluated in the SAR, as any leakage made possible by this change would fall within design specifications of maximum plant leakage. No new equipment challenges or malfunction consequences are introduced.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: The possibility of an accident of a different type than previously evaluated remains the same, as current accident evaluations consider collective leakage rates, which would include those pathways being introduced in this change.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: The possibility of a malfunction of equipment important to safety remains the same as previously evaluated in the SAR, as current equipment malfunction evaluations consider collective leakage rates, which would include any leakage from those pathways being introduced.

7. Will reduce the margin of safety as defined in the basis for any Technical Specification. ☐ Yes ☒ No

BASIS: The margin of safety for Containment leakage or pressure isolation valve leakage as defined in the Technical Specifications Basis will not be reduced. The testing programs which quantify the leakage for the specified limits are not being changed. Any leakage detected by increasing the acceptance criteria from "eliminated" to "as low as practical levels" would be evaluated to determine if the limits were affected. Therefore, the margin of safety as defined in the Technical Specifications will not be reduced.

III. Environmental Evaluation ☐ Not applicable per Environmental Evaluation Applicability Review

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

A. Environmental Protection Plan

1. Will require a change in the Environmental Protection Plan. ☐ Yes ☒ No

BASIS: The EPP does not address specific leakage pathways, and will not require change.

B. Unreviewed Environmental Question

1. Concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB.

   ☐ Yes ☒ No

   BASIS: The impact posed by effluent release is adequately evaluated in the FES, and the pathways introduced by this change will not affect the criteria used in determining any environmental impact.

2. Concerns a significant change in effluents or power level.

   ☐ Yes ☒ No

   BASIS: Maintaining all gas and water leakage to as low as practicable levels, as required by Tech Specs and NUREG - 0730, will not create a significant change in effluents, merely allow alternate pathways that provide those effluents.

3. Concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1 above, which may have a significant environmental impact.

   ☐ Yes ☒ No

   BASIS: Effluent leakage is reviewed and evaluated in the applicable documents.

Signatures and Approvals

Evaluated: Scott Lily 5/8/95
ORIGINATOR / DATE

Reviewed/Approved: 5/8/95
REVIEWER / DATE

Plant Safety Review Committee Review

Ricky L Patton 5/9/95
CHAIRMAN, PSRC / DATE

95-070 - P&SE

GRAND GULF NUCLEAR STATION UNIT 1
CHANGES, TESTS OR EXPERIMENTS SAFETY AND ENVIRONMENTAL EVALUATION FORM

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: D. R. Franklin DEPT/SECT: P&SE EVAL. #: 95-0057.R00

DOCUMENT EVALUATED: 03-1-01-6, Reactor Vessel In-Service Leak Test. Also reference WO 144875, WI&IR #2.

REFERENCES: Technical Specification Sections 3.4.6, 3.4.10, 3.10.1, 3.5.2, 6.4.2; FSAR Sections 5.2.5.2.k, 5.4.1.3, 5.4.7.1.2, 5.4.7.1.3, 5.4.7.2.7, 5.4.7.3.1, 5.4.5.4; GGNS-M-189.1, Pump and Valve Inservice Testing Program; 10 CFR 50.55a(c), 10 CFR 50, Appendix A, GDC 55

FSAR CHANGE REQUIRED? ☐ Yes ☑ No CR # N/A

FSAR SECTIONS TO BE REVISED: N/A

TRM CHANGE REQUIRED? ☐ Yes ☑ No

TECH. SPEC. CHANGE REQUIRED? ☐ Yes ☑ No CR # N/A

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? ☐ Yes ☑ No

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? ☐ Yes

THE RESPONSIBLE ORGANIZATIONS MUST BE NOTIFIED PRIOR TO IMPLEMENTING THIS CHANGE.

B. Executive Summary (ALSO SERVES AS INPUT TO NRC SUMMARY REPORT)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: To allow performing the Reactor Vessel In-Service Leak Test with the disc removed from 1E12F050B. The disc will be removed and the retaining plate/cap assembly will be installed as normal. Valve 1E12F053B, the other RCS PIV in series with 1E12F050B in the 12"-DBB-68 line, will become a test boundary valve. The 3/4"-DBB-66 test connection line from DBB-68 will also be pressurized and valve 1E12F058B will become a test boundary valve. Reference P&ID M-1085A.

REASON FOR CHANGE, TEST OR EXPERIMENT: This will allow the Reactor Vessel In-Service Leak Test to be performed while rework is being performed on the 1E12F050B disc per MNCR 0184-95. Reference WO 144875.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: Per Operability Revie / for MNCR 0184-95. 1E12F050B is declared inoperable until repaired and retested satisfactorily. This Safety Evaluation assumes 1E12F050B is inoperable and reviews only performing the Reactor Vessel In-Service Leak Test with disc removed. With the disc removed but the retainer plate/cap assembly installed as normal, the pressure integrity of 1E12F050B is still intact. However, without the disc, the normal test boundary must be moved upstream to the 1E12F053B and 1E12F058B valves. This change in test boundary does not create an unreviewed safety question as summarized in the Safety Evaluation Review.

II. Safety Evaluation

☐ Not applicable per Safety Evaluation Applicability Review

A. Technical Specifications

1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications.
   ☐ Yes
   ☑ No

   BASIS: The proposed change to perform the Reactor Vessel In-Service Leak Test with the disc removed from check valve 1E12F050B does not represent a change to the Technical Specifications. Check valve 1E12F050B is used to meet required actions for Tech Spec 3.4.6, however, this Tech Spec is not applicable in Mode 4. Tech Spec Sections 3.4.10, RHR Shutdown Cooling Requirements (also 3.10.1), and 3.5.2, ECCS Shutdown Limitations, are not being changed and still apply. Tech Spec 6.4.2 also does not change and 1E12F050B will still be maintained in accordance with the Inservice Testing Program.

B. Unreviewed Safety Question

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

1. May increase the probability of occurrence of an accident previously evaluated in the SAR.
   ☐ Yes
   ☑ No

   BASIS: Globe valve 1E12F053B will become a boundary valve for the Reactor Vessel In-Service Test. This valve already serves as one of the two RCS PIVs, in series with 1E12F050B. Valve 1E12F053B has been shown to be basically leak tight (3789 ml/min allowable leak rate, 50 ml/min actual) at a test differential pressure of 1050 psid per LLRT WO 140719. Also, 3/4"-DBB-66 and valve 1E12F058B will become part of the test boundary. The design rating of these components per MS-03 is more than adequate for the Reactor Vessel In-Service Leak Test conditions. There is another closed valve downstream of 1E12F058B which provides an additional test isolation backup to 1E12F058B. This is the 1E12F059B where the line terminates with a plugged 1-1/2" hose connection. For the reasons stated above, this change does not increase the probability of occurrence of an accident previously evaluated in the SAR.

2. May increase the consequences of an accident previously evaluated in the SAR.
   ☐ Yes
   ☑ No

BASIS: Performing the Reactor Vessel In-Service Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not change the intent or method of the Leak Test. The conditions (pressure and temperature) and flow path for the test remain basically the same. Valve 1E12F050B retains its pressure integrity with the disc removed since the cap assembly is a designed pressure boundary for the valve. The design ratings of the new test boundaries, 1E12F053B, 1E12F058B, and associated piping, are adequate for the test conditions. This change does not increase the consequences of an accident previously evaluated in the SAR.

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: Performing the Reactor Vessel In-Service Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not change the intent or method of the Leak Test. The 1E12F053B and 1E12F050B which will be subjected to reactor pressure will be in a passive state isolated in the closed position during the test. The 1E12F053B was local leak rate tested and was within the allowable leakage rate of 1 gpm per Tech Spec 3.4.6.1. The function of the 1E12F053B valve is to protect the low pressure RHR B piping from being over pressurized. The design ratings of the new test boundaries exceed test conditions. For these reasons this change does not increase the probability of occurrence of a malfunction of equipment.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: Performing the Reactor Vessel In-Service Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not change the intent or method of the Leak Test. The conditions and flow path for the test remain basically the same. The low pressure piping being protected via 1E12F053B has not been affected by this change, therefore, the consequences of the failure of 1E12F053B to protect the low pressure piping remains the same.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: Performing the Reactor Vessel In-Service Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not change the intent or method of the Leak Test. The conditions (temperature and pressure) and flow path for the test remain basically the same. This change does not increase the possibility for an accident of a different type than any previously evaluated in the SAR.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: Performing this Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not change the intent or method of the test. The conditions and flow path for the test remain basically the same. Subjecting the 1E12F053B to test pressure is well within the design and safety function of the valve to protect the low pressure piping and removing the disc from 1E12F050B does not affect the pressure boundary of the component, therefore, this change does not create the possibility for a malfunction of equipment of a different type than any previously evaluated in the SAR.

7 Will reduce the margin of safety as defined in the basis for any Technical Specification. ☐ Yes ☑ No

BASIS: Performing this Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not change the intent or method of the test. The conditions and flow path for the test remain basically the same. The margin of safety as defined in Tech Spec 3.4.6 requires RCS PIVs (1E12F053B & 1E12F050B) be operable in Modes 1, 2, and 3 and have leakage ≤ 1 gpm at 1050 ±10 psig. The Reactor Vessel In-Service Leak Test is performed in Mode 4. In Modes 1, 2 and 3, the loss of 1E12F050B would require isolation of the high pressure piping from the low pressure piping by isolating 1E12F053B. This intent is being met by the proposed test boundaries, therefore this change does not reduce the margin of safety for any Tech Spec.

III. Environmental Evaluation

☐ Not applicable per Environmental Evaluation Applicability Review

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

A Environmental Protection Plan

1 Will require a change in the Environmental Protection Plan. ☐ Yes ☑ No

BASIS: Performing the Reactor Vessel In-Service Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not require a change to the Environmental Protection Plan.

B Unreviewed Environmental Question

1 Concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB. ☐ Yes ☑ No

BASIS: Performing this Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B will not result in a significant increase in any adverse environmental impact.

2. Concerns a significant change in effluents or power level.

☐ Yes
☑ No

BASIS: Performing this Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B does not change the intent or method of the test. The conditions and flow path for the test remain basically the same. No change in effluents or power level will be required.

3. Concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1 above, which may have a significant environmental impact.

☐ Yes
☑ No

BASIS: Performing the Reactor Vessel In-Service Leak Test with the disc removed from check valve 1E12F050B and the test boundary changed to 1E12F053B and 1E12F058B will not result in a significant increase in any adverse environmental impact.

Signatures and Approvals

Evaluated: Dennis R. Franklin 5-22-95
ORIGINATOR / DATE

Reviewed/Approved: [signature] 5/22/95
REVIEWER / DATE

Plant Safety Review Committee Review

Ruby L Patton 5/22/95
CHAIRMAN, PSRC / DATE

95-074-P&SE

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: Alan J. Malone DEPT/SECT: P&SE EVAL. #: 95-0059-R00

DOCUMENT EVALUATED: Change to Technical Requirements Manual (TRM) Tables TR3.6.1.3.1-1 (Primary Containment Isolation Valves) and TR3.6.4.2-1 (Secondary Containment Isolation Valves)

REFERENCES: FSAR Section 6.2 & Chapter 15; EER 94/6182, with EERR dated 5/12/95; AECM-84/0330; MAEC-89/0124; Specification GGNS-M-189.1; 10 CFR 50.55a(f); ASME Code, Section XI

FSAR CHANGE REQUIRED? ☒ Yes ☐ No CR # ________

FSAR SECTIONS TO BE REVISED: N/A

TRM CHANGE REQUIRED? ☒ Yes ☐ No

TECH. SPEC. CHANGE REQUIRED? ☐ Yes ☒ No CR # N/A

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? ☐ Yes ☒ No

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? ☐ Yes

THE RESPONSIBLE ORGANIZATIONS MUST BE NOTIFIED PRIOR TO IMPLEMENTING THIS CHANGE.

B. Executive Summary (ALSO SERVES AS INPUT TO NRC SUMMARY REPORT)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: This Change revises the maximum allowable stroke times for various primary and secondary containment isolation valves listed in Technical Requirements Manual (TRM) Tables TR3.6.1.3.1-1 (Primary Containment Isolation Valves) and TR3.6.4.2-1 (Secondary Containment Isolation Valves). It does not revise times for drywell isolation valves or any other valves.

REASON FOR CHANGE, TEST OR EXPERIMENT: Engineering Evaluation Report (EER) No. 94/6182 was initiated because Specification GGNS-M-189.1, GGNS Unit 1 Pump and Valve Inservice Testing Program, Appendix A, "Bases for Maximum Stroke Times of Power Actuated Valves," implies that numerous valves in the inservice testing (IST) program have analytically-based maximum stroke time limits. Although some of the valves listed in Appendix A have paragraphs in the Safety Analysis Report (SAR) listed, for which explicit time limits are given, many of the valves are identified only as having limits in either GGNS Technical Specifications Table 3.6.4-1 for primary containment isolation valves or Table 3.6.6.2-1 for secondary containment isolation valves. Tables 3.6.4-1 and 3.6.6.2-1 have been relocated to the Technical Requirements Manual (TRM) as Tables TR3.6.1.3.1-1 and TR3.6.4.2-1, respectively.

B. Executive Summary (Continued):

REASON FOR CHANGE, TEST OR EXPERIMENT (CONTINUED): In "Detailed Description of Problem," the EER noted that many of the valves listed in TRM Tables TR3.6.1.3.1-1 and TR3.6.4.2-1 do not have analytical safety bases for the listed maximum allowable stroke times. In many cases, the listed maximum allowable stroke times for these valves had been arbitrarily determined based on either previous performance data or on information supplied by the manufacturer based on performance of typical valves of the same model/size.

Due to maintenance, design changes, and/or normal wear, performance of some valves has approached or exceeded limits in the TRM tables, resulting in additional man-hours and man-rem exposure to adjust their performance to again be within TRM limits.

In a response to EER 94/6182, Nuclear Plant Engineering (NPE) agreed that there were no technical bases for the stroke times for many of the valves in the TRM tables.

In the response, NPE identified, in an attached Table 1, valves with specific stroke time limits which were identified in the SAR. NPE also identified, in an attached Table 2, valves without specific limits clearly identified in the SAR but for which certain implied limits could be identified from the SAR based on certain analyzed events in which their positions were important. An example is the function of the Standby Gas Treatment System (SGTS) in drawing down the Auxiliary Building atmosphere, which requires that secondary containment isolation valves must be closed within the time limit for the SGTS to perform its function.

NPE's Response identified numerous valves with primary containment and secondary containment isolation functions for which the maximum stroke time limits in the TRM were shorter than could be justified by their primary or secondary containment isolation function. The valves considered in the TRM changes for which this Safety Evaluation is written are primary containment isolation valves with indirect pathways for leakage, for which the NPE Response identified a 60-second closing time limit, and secondary containment isolation valves, for which the NPE Response identified a 120-second closing time limit.

NPE's Response also identified drywell isolation valves whose closure time limits were not supported by the safety analysis. Additional analysis is required to identify the time limits for drywell isolation; therefore, these valves are not included in these TRM changes.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: The Safety Evaluation concludes that increasing the maximum allowable isolation times for the primary and secondary containment isolation valves identified in these TRM changes to the isolation times supported by the SAR safety analyses does not increase the probability or consequences of an accident or malfunction of equipment important to safety. The times currently listed for these valves in the TRM tables identified above do not have any safety significance to the plant.

Performance of inservice stroke testing of these valves will continue in accordance with the GGNS Unit 1 Pump & Valve Inservice Testing Program (Specification GGNS-M-189.1), as required by ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, "Inservice Inspection," which is referenced in and required by GGNS Technical Specifications 5.5.6 and TRM 7.6.3.3. This practice will minimize the possibility of operating with degraded components, which will prevent an increase in the probability of occurrence and the consequences of malfunctioning equipment.

II. Safety Evaluation

☐ Not applicable per Safety Evaluation Applicability Review

A. Technical Specifications

1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications. ☐ Yes ☒ No

BASIS: These changes do not change the requirements in GGNS Technical Specifications SR 3.6.1.3.4 and SR 3.6.4.2.2 to perform Inservice Testing (IST) exercising tests on the valves listed in Technical Requirements Manual (TRM) Tables TR3.6.1.3.1-1 (Primary Containment Isolation Valves) and TR3.6.4.2-1 (Secondary Containment Isolation Valves), nor do they change the actions to be taken if a valve fails to meet the maximum allowable stroke time listed in the TRM tables identified above.

In many cases, the listed maximum allowable stroke times for these valves had been arbitrarily determined based on either previous performance data or on information supplied by the manufacturer based on performance of typical valves of the same model/size.

These changes do not change any maximum allowable stroke times explicitly stated in any GGNS Technical Specification.

Neither the IST program requirements stated in GGNS Technical Specification 5.5.6 and TRM 7.6.3.3, nor the underlying IST program specified in ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, "Inservice Inspection," Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," 1980 Edition with Winter, 1980, Addenda, are being changed.

The maximum allowable stroke times currently in the TRM tables identified above were previously listed in GGNS Technical Specification Tables 3.6.4-1 and 3.6.6.2-1, but were relocated to the TRM in Amendment 102; therefore, changes to these tables do not require a Technical Specification change.

B. Unreviewed Safety Question

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

1. May increase the probability of occurrence of an accident previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: These changes do not affect the probability of occurrence of any accident evaluated in the SAR because they do not make any changes to the physical plant or any operating procedures identified in the SAR or Technical Specifications. The changes do not change the actual operating characteristics of any valve, nor do they affect the valve's performance in any way that could lead to an accident occurring. No physical or procedural changes in the plant are being made. These valves are primary and secondary containment isolation valves which are required to close to isolate primary or secondary containment under certain accident conditions. (Some of these valves may also have safety functions to open, but these TRM changes do not affect the opening time limits.)

1. May increase the probability of occurrence of an accident previously evaluated in the SAR **(Continued):**

BASIS (CONTINUED): Most of these valves are normally open during power operation and are required to close on primary and/or secondary containment isolation signals; however, some of them may be closed or opened infrequently during power operation.

To affect the probability of occurrence of an accident would require that one or more of these valves were in a position different from that analyzed in the accident scenario **at the onset of the accident sequence**. These TRM changes do not make any changes to the operating procedures or physical changes in the plant; therefore, they will not affect the initial position of any valve at the onset of any accident.

(Failure of a valve to respond to an isolation signal generated during the accident sequence or failure of the valve to close within the maximum closure time analyzed for the accident could certainly affect the consequences of the accident, but they do not affect the probability of occurrence. In addition, a valve which auto-isolated due to a malfunction of its control circuit could affect the probability of occurrence of an accident; however, such malfunctions are beyond the scope of this Safety Evaluation, since the proposed TRM changes do not affect such malfunctions.)

The probability of occurrence of an accident is affected by external factors not within the control of the Owner or Operator, as well as by internal design factors within the control of the Owner or Operator. However, in order to affect the probability of occurrence of an accident, the Owner would have to change the initial position of one or more of the valves identified in the TRM tables identified above. No changes to the operating procedures to change the operating positions of any valve are being made, and no physical changes in the plant are being made.

It is possible that the probability of occurrence of an accident could be increased if a valve's stroke time were decreased to the point that severe water hammer was generated by the closure of the valve. None of these TRM changes represent decreases in the stroke time limits; they are all increases in the limits.

Since the proposed changes do not change the operating position of any valve, they do not affect the probability of occurrence of any accident analyzed in the SAR.

2. May increase the consequences of an accident previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: These changes do not affect the consequences of any accident evaluated in the SAR, unless they would cause a required valve to fail to be in its safety position when it was required to be, or unless their speed of closure caused additional problems.

These changes do not make any changes to the physical plant or any operating or accident response procedures identified in the SAR or Technical Specifications; therefore, the ONLY way these changes could affect the consequences of an accident previously evaluated in the SAR is by increasing its allowable isolation time to the extent that the valve could still be closing when it is required to be fully closed or by causing damage to the plant or piping system by its closing speed.

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued)**:

BASIS (CONTINUED): All of the valves addressed in these changes are either primary containment isolation valves listed in TRM Table TR3.6.1.3.1-1 or secondary containment isolation valves listed in TRM Table TR3.6.4.2-1. No drywell isolation valves are included in the proposed TRM changes.

All of the valves addressed in these changes are designed to automatically isolate (move to the fully closed position) in response to an isolation signal. None of the isolation signals are affected by these changes, nor are the actions of the valves to close in response to the signals being changed. Although some of the valves may be capable of being reopened, or even stopped from fully closing by Operator action, such actions are not within the scope of the changes being made to the TRM tables listed in Section 1.A.

The ONLY possible ways that these changes could affect the consequences of any accident would be if one or more valves failed to isolate to its/their safety position within the time allowed by the accident analysis, or if the the closure of the valves in too fast a time caused water hammer that damaged the piping or supports further.

It is possible that the probability of occurrence of an accident could be increased if a valve's stroke time were decreased to the point that severe water hammer was generated by the closure of the valve. None of these TRM changes represent decreases in the stroke time limits; they are all increases in the limits.

The accident analyses in the SAR that require primary and secondary containment as part of the accident response are Loss of Coolant Accidents (LOCAs), which are described in SAR Section 6.2. The accident analyses assume that the primary and secondary containments are isolated within specified times after the start of the accidents in order to prevent radioactive releases to the environment in excess of the guidelines established in 10 CFR 100. Although some other accidents analyzed in SAR Chapter 15 expect or require isolation of some primary and/or secondary containment isolation valves (such as Main Steam Isolation Valves or other air-operated valves with fail-safe isolation functions), LOCAs are the limiting accidents for primary and secondary containment isolation valve closure.

As discussed in the Nuclear Plant Engineering (NPE) Response to Engineering Evaluation Request 94/6182, primary containment and secondary containment isolation valves may have either explicit or implicit analytical closure time limits based on the LOCA analyses, as well as for a variety of more limited events which require primary and/or secondary containment isolation, which are described in Chapter 15.

The NPE Response provided two lists of valves: Those with explicit analytical closure time limits (specific times stated in the SAR), which were described in Table 1 in the NPE Response, and those without explicit limits but for which isolation time limits are implied (implicit limits) from other events (e.g., SGTS drawdown) described in the SAR, which were described in Table 2 in the NPE Response.

None of the valves included in this TRM change have explicit limits listed in Table 1. They are all specifically identified in Table 2 as either primary containment isolation valves with indirect leak paths (for which the maximum analytical closure time is 60 seconds, as specified on Page 8 of the NPE Response) or secondary containment isolation valves (for which the maximum analytical closure time is 120 seconds, as also specified on Page 8 of the NPE Response).

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued)**:

BASIS (CONTINUED): The postulated radioactive release from the plant will be limited to less than the guidelines established in 10 CFR 100 if all primary and secondary penetrations are isolated in less than these analyzed times.

None of the valves included in this TRM change are drywell isolation valves, for which Table 2 of the NPE Response indicates additional analysis is required.

The consequences of an accident would be adversely affected if a valve malfunction caused by these TRM changes prevented the valve from closing within its required time. The probability of occurrence and consequences of equipment malfunctions are addressed in Sections II.B.3 and II.B.4 of this Safety Evaluation. These analyses conclude that neither the probability of occurrence nor consequences of equipment malfunctions would be increased by these TRM changes.

The EER Response from NPE specifies that primary containment isolation valves must close within 60 seconds to isolate penetrations which do not provide a direct pathway between containment and auxiliary building atmospheres, unless longer time limits have been specifically approved by the Nuclear Regulatory Commission (NRC). All of the primary containment valves identified in these TRM changes currently close in less than 30 seconds (half of the 60 second limit), except for four valves: 1E51F031, 1G41F028, 1G41F029, and 1G41F044, all four of which close within 45 seconds.

These four valves are motor-operated valves, which are highly predictable in their operating-time performance and have been shown over many years of operational performance to close reliably within a very narrow time band. Therefore, all of the primary containment isolation valves in these TRM changes can be expected to continue to close within the 60 second limit.

The EER Response from NPE also specifies that secondary containment isolation valves must close within 120 seconds, unless longer time limits have been specifically approved by the Nuclear Regulatory Commission (NRC). The 120 second isolation time is specified in SAR Section 6.2.3.1.1, which states that the secondary containment, in conjunction with the Standby Gas Treatment System (SGTS), is required to maintain a 1/4-inch water gauge (w.g.) negative pressure in the Auxiliary Building within 120 seconds after actuation. As described in FSAR Section 6.2.3.2, the SGTS is capable of maintaining the secondary containment negative pressure in spite of the failure of all nonqualified lines 2 inches and smaller or with the failure of a single nonisolated line as large as 4 inches.

All of the secondary containment valves identified in these TRM changes currently close in less than 60 seconds, except for eight Plant Service Water (PSW) System valves: 1P44F116 through 1P44F123.

The secondary containment valves (other than the eight PSW valves listed above) would not significantly affect the drawdown rate of the SGTS unless their operating times were increased to near the proposed limit of 120 seconds. They should not affect the ability of the SGTS to draw down the Auxiliary Building atmosphere for the following reasons:

1. They are easily capable of closing in less than 60 seconds and there are no plans to increase their normal closing times.

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued)**:

BASIS (CONTINUED):

2. The 120 second limit is intended to be used **only** for determining whether or not a valve should be declared Inoperable. It will not be the sole criterion for determining whether or not valve performance is Acceptable in accordance with ASME Code, Section XI.

3. ASME Code, Section XI, specifies that corrective action is required if a valve exceeds its limiting value of full stroke time. In accordance with the GGNS Unit 1 Pump and Valve IST Program, Specification GGNS-M-189.1, that limiting value of full-stroke time is established as a multiple of the average stroke time performance of the valve. Currently, the multiple is 1.35 for valves with full stroke times greater than 10 seconds.

4. Since they close in less than 60 seconds, their limiting values of closing time are limited to less than 1.35 × 60 seconds, or less than 81 seconds. This is well below the 120 second limit for Auxiliary Building drawdown.

Therefore, all of the secondary containment isolation valves identified in these TRM changes, except possibly the eight PSW valves identified above, will continue to close well within the 120 second limit, even with these TRM changes, and will not affect the consequences of an accident requiring secondary containment integrity.

The eight PSW valves identified above are the only secondary containment isolation valves which take longer than 60 seconds to close. These valves currently close within a relatively narrow band of 90 to 100 seconds. The 90-second limit at the low end of the band is specified to minimize water hammer when they close. These valves are equipped with air actuators which fail closed on loss of air pressure. Depending on their conditions, one or more of these valves **could** fail to close in time to permit the Standby Gas Treatment System (SGTS) to draw down the Auxiliary Building atmosphere to 1/4-inch w.g. within 120 seconds.

It is unlikely that increasing the closing time limit for one or more of the eight PSW Valves identified above to 120 seconds would cause the SGTS to fail to draw down the Auxiliary Building atmosphere to 1/4-inch w.g. within 120 seconds for the following reasons:

1. They are all in water-filled penetrations; therefore, they are not exposed to Auxiliary Building atmosphere at any time, except as a possible result of pipe rupture inside the Auxiliary Building.

2. They all have air-operated fail-safe actuators, which cause the valves to automatically close on loss of air pressure or electrical power. They do not require power to close and, therefore, except due to component malfunction, are postulated to close on a secondary containment isolation signal. Since more than one independent component failure in addition to the initiating event is not a credible occurrence, all but one of these eight valves can be postulated to close during the accident.

3. All of the PSW piping isolated by these valves have redundant isolation valves; that is, there are two of these valves on each PSW pipe which penetrates the Auxiliary Building. In accordance with Point 2 above, which postulates no more than one independent component failure in addition to the initiating event, each PSW pipe can be reliably expected to be isolated by at least one valve during an accident requiring secondary containment.

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued)**:

BASIS (CONTINUED):

4. The PSW piping outside the Auxiliary Building is buried several feet underground. Although the piping outside the Auxiliary Building isolation boundary is not safety-related, the interior of the pipe would not be exposed to the atmosphere, even if it ruptured, until the water in the pipe drained out and a path was opened up to the atmosphere.

5. Since it is buried, the water cannot readily drain out of the pipe. Following an accident, the water would be most likely to drain out the ends (at the radial wells or inside the Auxiliary Building). This draining would be a slow process for the reasons discussed in the following points.

6. At the radial wells, check valves at the pumps would retard drainage back through the pump, even if the pumps were not running. Even if the piping in the Auxiliary Building ruptured, it would take a long time (several minutes) for the piping between the Auxiliary Building and the radial well pumps (over a mile of pipe) to drain so that the atmospheres would be exposed.

7. Even if the piping outside the Auxiliary Building ruptured, the piping inside the Auxiliary Building would also have to rupture in order to expose the isolation valve discs to the Auxiliary Building atmosphere. While this piping is not safety-related, it is constructed to the Power Piping Code, ASME B31.1, and, therefore, can be expected to remain reasonably intact. Although portions may crack during a seismic event, it can reasonably be expected, based on industry experience during other seismic events with piping constructed to the same Code, to remain in one piece and supported. This means the water would not instantaneously run out of the pipe and expose the isolation valve discs. It would, instead, probably maintain the piping water-filled for most of the time that the valves were closing.

8. Even if the piping outside the Auxiliary Building ruptured and the piping inside the Auxiliary Building ruptured, the valve discs would not be exposed to atmospheres inside and outside the Auxiliary Building until water drained out of the pipe sufficiently to expose a leakage pathway. During the first few minutes after an accident, there is no significant pressure inside primary containment; therefore, assuming a containment penetration were open to atmosphere, there would not be any significant pressure in the Auxiliary Building. Some significant driving pressure inside the Auxiliary Building would be required in order to push the water in the PSW piping out of the underground piping. Without a driving pressure head, the water would drain out slowly, but it would take significantly longer that two minutes to drain the piping completely.

Based on the above, even if the maximum allowable closing times for the eight PSW valves listed above were increased to 120 seconds, it would not affect the ability of the SGTS to draw down the Auxiliary Building atmosphere to 1/4-inch w.g.

Since the proposed changes do not exceed the limits in Table 2 of the NPE Response and they would not increase the probability of occurrence or consequences of equipment malfunctions, they do not affect the consequences of any accident analyzed in the SAR.

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: These changes do not affect the probability of occurrence of a malfunction of any equipment important to safety previously evaluated in the SAR because they do not make any changes to the physical plant or any operating or accident response procedures identified in the SAR or Technical Specifications and they do not affect programs already in place to prevent the valves from operating under degraded conditions.

The probability of a malfunction of equipment important to safety is affected by changes in the conditions under which it operates (environment, power supply inputs, mechanical and electrical condition, etc.), none of which are being changed by these changes. No physical or procedural changes in the plant are being made.

The probability of malfunction of equipment important to safety may also be increased by operating the equipment under degraded conditions for long periods. Such degraded conditions may include under-voltage, under-lubrication, reduced air supply and/or pressure, etc. One of the symptoms of operation under degraded conditions, at least for some valves, is an increase in the measured stroke times. To the extent that increasing the allowable maximum stroke time limit may allow a valve to be operated under degraded conditions for extended periods, these increases in the limits might increase the probability of a malfunction of the affected valves, if they were the only applicable limits.

However, the TRM maximum allowable stroke time limit is not the only stroke time limit for these valves. All of these valves are also required to be exercised and stroke time tested in accordance with ASME Code, Section XI, Subsection IWV, as required by and specified in GGNS Technical Specification 5.5.6, TRM 7.6.3.3, and 10 CFR 50.55a(f).

The Nuclear Regulatory Commission (NRC) provided guidance and clarification on the performance of IST on pumps and valves in accordance with Section XI with the issuance of Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs" (MAEC-89/0124). In Attachment 1 to GL 89-04, Section 5, entitled "Limiting Values of Full-Stroke Times for Power Operated Valves," the NRC stated the following guidance:

> "The purpose of the limiting value of full-stroke time is to establish a value for taking corrective action on a degraded valve before the valve reaches the point where there is a high probability of failure to perform its safety function if called upon. The NRC has, therefore, established the guidelines described below regarding limiting values of full-stroke time for power operated valves.
>
> "The limiting value of full-stroke time should be based on the valve reference or average stroke time of a valve when it is known to be in good condition and operating properly. The limiting value should be a reasonable deviation from this reference stroke time based on the valve size, valve type, and actuator type. The deviation should not be so restrictive that it results in a valve being declared inoperable due to reasonable stroke time variations. However, the deviation used to establish the limit should be such that corrective action would be taken for a valve that may not perform its intended function."

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR **(Continued):**

BASIS (CONTINUED): The GGNS Unit 1 Pump and Valve Inservice Testing Program, Specification GGNS-M-189.1, "GGNS Unit 1 Pump and Valve Inservice Testing Program," has been written to comply with ASME Code, Section XI, requirements, including specific requirements in GGNS Technical Specifications and the TRM. The NRC guideline stated above is implemented in Specification GGNS-M-189.1, Appendix C, entitled "Calculation of IST Maximum Stroke Times." Appendix C describes the method of calculating a maximum stroke time based on the performance of the valve. Paragraph 7.1.3 of Appendix C incorporates the guidance in the section of GL 89-04 quoted above.

In essence, Appendix C requires that, if a limit calculated based on the performance of the valve is less than an analytical limit in the SAR, Technical Specifications or TRM, the lower limit shall be the limit for taking corrective action, before the valve performance degrades to the point that a malfunction could occur. If the analytical limit is less, the analytical limit is the limit for taking corrective action. Either limit minimizes the probability of operating the plant with valves in degraded condition.

In addition, it is possible that the probability of a component malfunction could be increased it a valve's stroke time were decreased to the point that severe water hammer was generated by the closure of the valve. None of these TRM changes represent decreases in the stroke time limits; they are all increases in the limits.

Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: These changes do not affect the consequences of a malfunction of any equipment important to safety previously evaluated in the SAR because they do not make any changes to the physical plant or any operating or accident response procedures identified in the SAR or Technical Specifications, and because existing inservice testing (IST) programs in accordance with ASME Code, Section XI, will minimize the likelihood of operating valves under degraded conditions.

The consequences of a malfunction of equipment important to safety could be affected by changes to the conditions under which it normally operates (environment, power inputs, mechanical and electrical condition, etc.), none of which are being changed by these changes. No physical changes to the plant or its components are being made, and no procedural changes in operation of the plant or response to any accident are being made.

The consequences of a malfunction of equipment important to safety may also be increased by operating the equipment under degraded conditions for long periods of time. Such degraded conditions may include under-voltage, under-lubrication, reduced air supply and/or pressure, etc.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR **(Continued)**:

BASIS (CONTINUED): One of the symptoms of operation under degraded conditions, at least for some valves, is an increase in the measured stroke times. To the extent that increasing the allowable maximum stroke time limit may allow a valve to be operated under degraded conditions for extended periods, these increases in the limits might increase the consequences of a malfunction of the affected valves, if they were the only applicable limits.

However, the TRM maximum allowable stroke time limit is not the only stroke time limit for these valves. All of these valves are also required to be exercised and stroke time tested in accordance with ASME Code, Section XI, Subsection IWV, as required and specified in GGNS Technical Specification 5.5.6, TRM 7.6.3.3, and 10 CFR 50.55a(g).

The NRC provided guidance and clarification on the performance of IST on pumps and valves in accordance Section XI by issuing Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs" (MAEC-89/0124). In Attachment 1 to Generic Letter 89-04, Section 5, entitled "Limiting Values of Full-Stroke Times for Power Operated Valves," the NRC provided guidance, which is quoted in Section II.B.3 of this Safety Evaluation.

The discussion in Section II.B.3 of this Safety Evaluation explained that additional limits imposed by ASME Code, Section XI, and the NRC guidance in Generic Letter 89-04, which are implemented in Specification GGNS-M-189.1, would minimize operating valves under degraded conditions.

In essence, Appendix C of Specification GGNS-M-189.1 requires that, if a limit calculated based on the performance of the valve is less than an analytical limit in the SAR, Technical Specifications or TRM, the lower limit shall be the limit for taking corrective action, before the valve performance degrades to the point that a malfunction could occur. If the analytical limit is less, the analytical limit is the limit for taking corrective action. Either limit minimizes the probability of operating the plant with valves in degraded condition.

In addition, it is possible that the consequences of a component malfunction could be increased if a valve's stroke time were decreased to the point that severe water hammer was generated by the closure of the valve. None of these TRM changes represent decreases in the stroke time limits; they are all increases in the limits.

Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be increased.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: These changes do not increase the possibility for an accident of a different type than any previously evaluated in the SAR because they do not make any changes to the physical plant or any operating or accident response procedures identified in the SAR or Technical Specifications, and because existing inservice testing (IST) programs in accordance with ASME Code, Section XI, will minimize the likelihood of operating valves under degraded conditions.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR **(Continued):**

BASIS (CONTINUED): To affect the possibility of a new accident would require that one or more of these valves were in a position different from its analyzed position **at the onset of the accident sequence** or that the valve failed to stroke to its safety position during its response to an isolation signal.

The possibility a new accident could be affected by external factors not within the control of the Owner or Operator, as well as by internal design factors within the control of the Owner or Operator. However, in order to affect the possibility of such an accident, the Owner would have to change the initial position of one or more of the valves identified in the TRM tables identified above. No changes to the operating procedures to change the operating positions of any valve are being made, and no physical changes in the plant are being made.

The ONLY possible ways that these changes could increase the possibility of any new accident would be if the valve failed to isolate to its safety position within the time allowed by current accident analyses or if changes in the valve stroke times introduced new instabilities, such as water hammer.

The probability of valves failing to move to their safety positions has been discussed in Section II.B.2 of this Safety Evaluation. The discussion concluded that these TRM changes would not increase the likelihood that the valves would not move to their safety positions. In addition, the discussion in Section II.B.3 of this Safety Evaluation explained that additional stroke time limits imposed by ASME Code, Section XI, and Generic Letter 89-04 would minimize operating valves under degraded conditions, which could also prevent the valves from closing when required.

The possibility for an accident of a different type could be increased by introducing instabilities into the system when the valves close, such as due to water hammer. Water hammer occurs when water flow is suddenly altered by stopping, starting or changing the direction of the water flow. The only way water hammer or other flow instabilities could be introduced by these TRM changes would be if a valve's closing stroke time were shortened to the point that severe water hammer or other flow instabilities were generated by the closure of the valve.

None of the TRM changes analyzed in this Safety Evaluation represent decreases in the stroke time limits; all of the changes are increases in the limits.

Therefore, the possibility for an accident of a different type than any previously evaluated in the SAR will not be increased.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: These changes do not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR because they do not make any changes to the physical plant or any operating or accident response procedures identified in the SAR or Technical Specifications.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR**(Continued)**:

BASIS (CONTINUED): The possibility of a new malfunction of equipment important to safety is affected by changes to its operating conditions (environment, power inputs, mechanical and electrical condition, etc.), none of which are being changed by these changes. No physical or procedural changes in the plant are being made.

The possibility of a new malfunction of equipment important to safety may also be increased by operating the equipment under degraded conditions for long periods. Such degraded conditions may include under-voltage, under-lubrication, reduced air supply and/or pressure, etc. One of the symptoms of operation under degraded conditions, at least for some valves, is an increase in the measured stroke times. To the extent that increasing the allowable maximum stroke time limit may allow a valve to be operated under degraded conditions for extended periods, these increases in the limits may increase the probability of a malfunction of the affected valves.

However, the TRM maximum allowable stroke time limit is not the only stroke time limit for these valves. All of these valves are also required to be exercised and stroke time tested in accordance with ASME Code, Section XI, Subsection IWV, as required and specified in GGNS Technical Specification 5.5.6, TRM 7.6.3.3, and 10 CFR 50.55a(f).

The NRC provided guidance and clarification on the performance of IST on pumps and valves in accordance with Section XI by issuing Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs" (MAEC-89/0124). In Attachment 1 to Generic Letter 89-04, Section 5, entitled "Limiting Values of Full-Stroke Times for Power Operated Valves," the NRC provided guidance, which is quoted in Section II.B.3 of this Safety Evaluation. The discussion in Section II.B.3 of this Safety Evaluation explained that additional limits imposed by ASME Code, Section XI, and the NRC guidance in Generic Letter 89-04, which are implemented in Specification GGNS-M-189.1, would minimize operating valves under degraded conditions, which could also prevent the valves from closing when required.

In addition, the possibility for an equipment malfunction could be increased by introducing instabilities into the system when the valves close, such as due to water hammer. Water hammer occurs when water flow is suddenly altered by stopping, starting or changing the direction of the water flow. The only way water hammer or other flow instabilities could be introduced by these TRM changes would be if a valve's closing stroke time were shortened to the point that severe water hammer or other flow instabilities were generated by the closure of the valve.

None of the TRM changes analyzed in this Safety Evaluation represent decreases in the stroke time limits; all of the changes are increases in the limits.

Therefore, the possibility of a malfunction of equipment important to safety different from any previously evaluated in the SAR will not be increased.

7. Will reduce the margin of safety as defined in the basis for any Technical Specification.

☐ Yes
☒ No

BASIS: These changes do not reduce th_ margin of safety as defined in the basis for any Technical Specification because they d_ not make any changes to the physical plant or any operating or accident response procedu_es identified in the SAR or Technical Specifications.

Margins of safety are associated with _he redundancy of two independent primary containment isolation valves, for primary contair_nent isolation valves, and with the redundancy of the Standby Gas Treatment System (SG_TS) as a backup for the secondary containment isolation valves.

In order to prevent reducing the mar_ins of safety associated with both primary and secondary containment isolation valves, the valves' capability of closing within specified maximum time limits when required must not be degraded. Such degraded conditions may include under-voltage, under-lubrication, reduced air supply and/or pressure, etc.

One of the symptoms of operation under degraded conditions, at least for some valves, is an increase in the measured stroke times. To the extent that increasing the allowable maximum stroke time limit may allow a valve to be operated under degraded conditions for extended periods, the margins of safety may be reduced.

However, the TRM maximum allowable stroke time limit is not the only stroke time limit for these valves. All of these valves are also required to be exercised and stroke time tested in accordance with ASME Code, Section XI, Subsection IWV, as required and specified in GGNS Technical Specification 5.5.6, TRM 7.6.3.3, and 10 CFR 50.55a(f).

The NRC provided guidance and clarification on the performance of IST on pumps and valves in accordance with Section XI by issuing Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs" (MAEC-89/0124).

In Attachment 1 to Generic Letter 89-04, Section 5, entitled "Limiting Values of Full-Stroke Times for Power Operated Valves," the NRC provided guidance, which is quoted in Section II.B.3 of this Safety Evaluation. The discussion in Section II.B.3 of this Safety Evaluation explained that additional limits imposed by ASME Code, Section XI, and the NRC guidance in Generic Letter 89-04, which are implemented in Specification GGNS-M-189.1, would minimize operating valves under degraded conditions, which could also prevent the valves from closing when required.

In addition, the margin of safety could be reduced by introducing instabilities into the system when the valves close, such as due to water hammer. Water hammer occurs when water flow is suddenly altered by stopping, starting or changing the direction of the water flow. The only way water hammer or other flow instabilities could be introduced by these TRM changes would be if a valve's closing stroke time were shortened to the point that severe water hammer or other flow instabilities were generated by the closure of the valve.

None of the TRM changes analyzed in this Safety Evaluation represent decreases in the stroke time limits; all of the changes are increases in the limits.

Therefore, the margin of safety as implicitly defined in the bases for Technical Specifications 3.6.1.3.1 and 3.6.4.2 will not be reduced.

III. Environmental Evaluation

☐ Not applicable per Environmental Evaluation Applicability Review

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

A. Environmental Protection Plan

1. Will require a change in the Environmental Protection Plan.

☐ Yes
☒ No

BASIS: Valve maximum allowable stroke times are not addressed in the Environmental Protection Plan.

B. Unreviewed Environmental Question

1. Concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB.

☐ Yes
☒ No

BASIS: As long as the primary and secondary isolation valves close within the maximum allowable stroke times addressed in the SAR, potential radioactive releases to the environment as a result of accidents will not exceed the guidelines of 10 CFR 100, and there will be no negative effect on the environment.

The proposed maximum allowable stroke times are in accordance with the times analyzed per NPE Response to EER 94/6182, which are intended to provide primary and secondary containment isolation sufficiently quickly after an accident that radioactive releases to the environment as a result of accidents will not exceed the guidelines of 10 CFR 100.

These changes do not change any physical plant system or component, nor do they change any operating or accident response procedures described in the SAR or Technical Specifications.

Since the maximum allowable stroke times addressed in these changes are within those addressed in the SAR, there will be no effect on the environment.

2. Concerns a significant change in effluents or power level..

☐ Yes
☒ No

BASIS: The proposed changes to the TRM maximum allowable stroke times of primary and secondary containment isolation valves do not represent changes in effluents or power level. They do not represent any physical changes in the plant or in any plant operating procedure that could affect effluents or power level. Discussion under Sections II.B.1 through II.B.4 above clearly explain why the probability of occurrence and consequences of accidents and equipment malfunctions, which could increase effluents, will not be increased.

3. Concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1 above, which may have a significant environmental impact.

☐ Yes
☒ No

BASIS: As long as the primary and secondary isolation valves close within the maximum allowable stroke times addressed in the SAR, potential radioactive releases to the environment as a result of accidents will not exceed the guidelines of 10 CFR 100, and there will be no negative effect on the environment.

The proposed maximum allowable stroke times are in accordance with the times analyzed per NPE Response to EER 94/6182, which are intended to provide primary and secondary containment isolation sufficiently quickly after an accident that radioactive releases to the environment as a result of accidents will not exceed the guidelines of 10 CFR 100.

These changes do no change any physical plant system or component, nor do they change any operating or accident response procedures described in the SAR or Technical Specifications.

Since the maximum allowable stroke times addressed in these changes are within those addressed in the SAR, there will be no effect on the environment.

Signatures and Approvals

Evaluated: A. J. Malone / 6-8-95
ORIGINATOR / DATE

Reviewed/Approved: [signature] 6/8/95
REVIEWER / DATE

Plant Safety Review Committee Review

Ricky L Patterson 6/15/95
CHAIRMAN, PSRC / DATE

95-075-P&SE

GRAND GULF NUCLEAR STATION UNIT 1
CHANGES, TESTS OR EXPERIMENTS SAFETY AND ENVIRONMENTAL EVALUATION FORM

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: K.L. Walker DEPT/SECT: P&SE/RE EVAL. #: 95-0060-R00

DOCUMENT EVALUATED: Technical Requirements Manual 7.7.1.1

REFERENCES: OQAM 3.4.2

FSAR CHANGE REQUIRED? ☑ Yes ☐ No CR # CR 95-043

FSAR SECTIONS TO BE REVISED: Appendix 3A, page 1.16-1

TRM CHANGE REQUIRED? ☑ Yes ☐ No

TECH. SPEC. CHANGE REQUIRED? ☐ Yes ☑ No CR # (n/a)

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? ☐ Yes ☑ No

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? ☐ Yes

THE RESPONSIBLE ORGANIZATIONS MUST BE NOTIFIED PRIOR TO IMPLEMENTING THIS CHANGE.

B. Executive Summary (ALSO SERVES AS INPUT TO NRC SUMMARY REPORT)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: This change removes the commitment to submit a summary Startup Report to the NRC under certain conditions. This requirement is contained in Regulatory Guide 1.16, C.1.a, and appears in the Technical Requirements Manual 7.7.1.1 as well as in the UFSAR Reg. Guide commitments listing. The report has typically been submitted following each reload, and this would no longer be required.

REASON FOR CHANGE, TEST OR EXPERIMENT: The report contains only basic information and references other programs and reports available on site. NRC approval of startup tests is not required, and complete information is readily available in plant records of the various tests performed after each reload. Adequate programs are in place to ensure that testing is done and that results are analyzed and screened for non-conformances or other problems. Summarizing this information for review and filing by the NRC does not enhance plant safety. It does, however, require use of valuable plant time and resources.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: This proposed change serves only to remove an administrative requirement associated with commitment to RG 1.16 C.1.a. Elimination of the summary Startup Test Report currently sent to the NRC after each reload will in no way increase the probability or consequences of accidents or malfunctions previously evaluated. NRC approval of startup test results is not required by RG 1.16, and the report is submitted for information purposes only. No evidence could be found that the Commission in any way based the GGNS Safety Evaluation Report on a requirement to submit startup test results following each reload. No modifications to the facility or to the conduct or review of startup testing will be done under this change. No new types of events could be created by elimination of this report, nor is any margin of safety affected. Thus elimination of the TRM requirement for the summary Startup Test Report and removal of the UFSAR commitment to comply with this aspect of RG 1.16 do not present an unreviewed safety question.

II. Safety Evaluation ☐ Not applicable per Safety Evaluation Applicability Review

A. Technical Specifications

1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications. ☐ Yes ☑ No

BASIS: Reporting requirements are contained in the TRM, Section 7.7, and are not described in the Technical Specifications.

B. Unreviewed Safety Question

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

1. May increase the probability of occurrence of an accident previously evaluated in the SAR. ☐ Yes ☑ No

BASIS: The proposed change makes no physical modifications to the facility or any operating, maintenance, or testing practices. Startup testing will continue to be conducted in accordance with applicable plant programs and Technical Specification requirements. Reporting or not reporting these results is unrelated to accident probability.

2. May increase the consequences of an accident previously evaluated in the SAR. ☐ Yes ☑ No

BASIS: The proposed change makes no physical modifications to the facility or any operating, maintenance, or testing practices. Startup testing will continue to be conducted in accordance with applicable plant programs and Technical Specification requirements. Reporting or not reporting these results is unrelated to accident consequences.

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: No modifications to any systems, structures, or components are being made by the proposed change. The change only removes the requirement to summarize startup test results to the NRC. Startup tests will continue to be performed and evaluated in accordance with applicable programs and requirements, and results will be available on-site for review at any time by the Commission. Therefore, there is no increase in the probability of malfunction of plant equipment.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: No modifications to any systems, structures, or components are being made by the proposed change. The change only removes the requirement to summarize startup test results to the NRC. Startup tests will continue to be performed and evaluated in accordance with applicable programs and requirements, and results will be available on-site for review at any time by the Commission. Therefore, there is no increase in the consequences of malfunction of plant equipment.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: No new accident possibilities are created since there are no physical modifications being made by the proposed change, nor are there any changes to the way testing is conducted or evaluated. Adequate plant programs and procedures, combined with Technical Specification surveillance requirements are in place to ensure that thorough startup testing is performed to confirm design predictions. NRC approval of startup tests is not required. The proposed change only affects reporting requirements for test results provided for NRC information purposes only.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

☐ Yes
☑ No

BASIS: No new equipment malfunction possibilities are created since there are no physical modifications being made by the proposed change, nor are there any changes to the way testing is conducted or evaluated. Adequate plant programs and procedures, combined with Technical Specification surveillance requirements are in place to ensure that thorough startup testing is performed to confirm design predictions. The proposed change only affects reporting requirements for test results provided for NRC information purposes only.

7. Will reduce the margin of safety as defined in the basis for any Technical Specification.

☐ Yes
☑ No

BASIS: There is no margin of safety related to reporting of information to the NRC. Commission approval of test results is not required, and adequate review of startup tests is provided for under existing plant programs and Technical Specifications. No change is being made to the startup test procedures or review processes. There is no mention in the Safety Evaluation Report of requiring startup test reporting to the NRC following reloads. Thus, removal of information-only reporting of startup tests results does not reduce the margin of safety.

III. Environmental Evaluation ☐ Not applicable per Environmental Evaluation Applicability Review

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

A. Environmental Protection Plan

1. Will require a change in the Environmental Protection Plan. ☐ Yes ☑ No

BASIS: Reporting requirements for startup testing are not addressed in the EPP.

B. Unreviewed Environmental Question

1. Concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB. ☐ Yes ☑ No

BASIS: Reporting requirements for startup testing are not addressed in the FES.

2. Concerns a significant change in effluents or power level.. ☐ Yes ☑ No

BASIS: The proposed change deals only with NRC administrative reporting requirements for startup test results and has no impact on effluents or power level. Any proposed changes in effluents or power level are processed in accordance with applicable plant programs which are not being changed.

3. Concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1 above, which may have a significant environmental impact. ☐ Yes ☑ No

BASIS: The proposed change deals only with NRC administrative reporting requirements for startup test results and has no impact on the environment.

Signatures and Approvals

Evaluated: Ken L. Walker 5/30/95
ORIGINATOR / DATE

Reviewed/Approved: Valerie W. Dunning Jr. 6-6-95
REVIEWER / DATE

Plant Safety Review Committee Review

6/29/95
CHAIRMAN, PSRC / DATE

95-077-PSE

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: Alan J. Malone DEPT/SECT: P&SE EVAL. #: 95-0075-R00

DOCUMENT EVALUATED: Changes to UFSAR Tables 6.2-44, 6.2-49 and 16B-3.6.4-1, and Technical Requirements Manual (TRM) Table TR3.6.1.3-1

REFERENCES: 10 CFR 50, Appendix J; UFSAR Section 6.2 and Chapter 15; ANSI/ANS 56.8-1987; UFSAR Change Request PL-93-007; GIN-95/01815; NUMARC, T. E. Tipton, letter, dated January 17, 1992, Subject: "NRC Proposed Revision to 10 CFR 50, Appendix J, Containment Leak Rate Testing"; Safety Evaluation 93-0049-R00

FSAR CHANGE REQUIRED? ☒ Yes ☐ No CR# 95-065

FSAR SECTIONS TO BE REVISED: Table 6.2-44; Table 6.2-49; and Table 16B-3.6.4-1

TRM CHANGE REQUIRED? ☒ Yes ☐ No

TECH. SPEC. CHANGE REQUIRED? ☐ Yes ☒ No CR# N/A

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? ☐ Yes ☒ No

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? ☐ Yes

THE RESPONSIBLE ORGANIZATIONS MUST BE NOTIFIED PRIOR TO IMPLEMENTING THIS CHANGE.

B. Executive Summary (ALSO SERVES AS INPUT TO NRC SUMMARY REPORT)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: This change deletes the requirements to perform Type C local leak rate testing on nine test connection valves listed in UFSAR Tables 6.2-44 and 6.2-49 and in Technical Requirements Manual (TRM) Table TR3.6.1.3-1 (formerly Tech Spec Table 3.6.4-1). The nine valves are the following:

Valve No.	Penetration No.	Penetration Service
1B21F025A	5	Main Steam A
1B21F025B	6	Main Steam B
1B21F025C	7	Main Steam C
1B21F025D	8	Main Steam D
1B21F030A and F063A	9	Feedwater A
1B21F030B and F063B	10	Feedwater B
1E51F072	17	RCIC Steam Supply

Various directives associated with the local leak rate testing program will also be revised as a result of this change.

B. Executive Summary (Continued):

REASON FOR CHANGE, TEST OR EXPERIMENT: The test connection valves are not required to be Type C local leak rate tested because they do not conform to the characteristics of valves that are required to be Type C tested under the definition of "Type C Test" as defined in 10 CFR 50, Appendix J, Definition II.H. They are small manual valves, are locked in closed position during power operation, are operated infrequently, and are not capable of remote or automatic operation. In addition, because these test connection pipes attach to their process pipes between inboard and outboard main isolation valves and have additional valves and pipe caps in series, these penetrations present multiple independent barriers to leakage through the penetration.

Eliminating the Type C tests of these test connection valves will save significant outage time, man-hours and man-rem exposure.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: The Safety Evaluation concludes that neither the probability nor the consequences of an accident or malfunction of equipment will be increased by exempting the local leak rate testing. Local leak rate testing is not effective in detecting mispositioned valves, which is the only likely reason that the probability of an accident or malfunction of equipment would be increased. The consequences of an accident or malfunction of equipment are minimized by the valves' construction, their infrequent operation, and administrative controls on the valves' disk positions.

Although these valves seem to be specifically identified in Appendix J, Definition II.H, as requiring Type C testing, this evaluation justifies why they should be exempted. In addition, GIN-95/01815 documents Nuclear Safety & Regulatory Affairs (NS&RA) position that these changes may be made under the provisions of the 10 CFR 50.59 program.

II. Safety Evaluation

☐ Not applicable per Safety Evaluation Applicability Review

A. Technical Specifications

1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications. ☐ Yes ☒ No

BASIS: Valves and other containment penetrations that must be Type B or C tested per 10 CFR 50, Appendix J, are listed in Technical Requirements Manual (TRM) Table TR3.6.1.3-1. This table was previously Tech Spec Table 3.6.4-1 but was removed from the Technical Specifications (TS) in Amendment 102. Part of the basis for removal of this table from the Technical Specifications, as stated in Generic Letter 91-08 and the Nuclear Regulatory Commission's (NRC's) Safety Evaluation of Tech Spec Amendment 102, is to "allow licensees to make corrections and updates to the list of components for which these TS requirements apply, under the provisions that control changes to plant procedures as specified in the Administrative Controls Section of the TS." Changes to TRM tables do not require a change to the Technical Specifications, although they do require compliance with the Administrative Controls Section of the Technical Specifications.

II. Safety Evaluation

☐ Not applicable per Safety Evaluation Applicability Review

B. Unreviewed Safety Question

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

1. May increase the probability of occurrence of an accident previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: The Appendix J leakage testing requirements are intended only to minimize leakage of radioactivity from the containment during and following an accident. Neither these changes nor the underlying Appendix J testing requirements affect the probability that an accident will occur. The changes do not affect the physical design, or operating position or status of any plant components or systems.

2. May increase the consequences of an accident previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: The response of the plant and its equipment to an accident requiring containment isolation is based on meeting the design and testing requirements in regulatory documents such as 10 CFR 50, Appendices A and J. Appendix J requires a program to be developed for conducting Type A, B and C tests.

10 CFR 50, Appendix J, Definition II.H. defines Type C test as follows:

> "H. 'Type C Tests' means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:
> "1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves;
> "2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;
> "3. Are required to operate intermittently under post-accident conditions; and
> "4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors."

Containment isolation valves are defined in Appendix J, Definition II.B., which reads as follows:

> "B. 'Containment isolation valve' means any valve which is relied upon to perform a containment isolation function."

For the rest of this justification I will refer to the four conditions, given in Definition II.H., that require containment isolation valves to be Type C leak rate tested as **Qualifiers 1, 2, 3 and 4**.

"Containment isolation function" is not specifically defined in Appendix J; however, 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, General Design Criteria (GDC) 54 through 57 contain the design requirements for piping systems penetrating primary containment. GDC 55, 56 and 57, in particular, describe the number, locations and arrangement of valves which perform the containment isolation function.

2. May increase the consequences of an accident previously evaluated in the SAR (**Continued**):

BASIS (CONTINUED): Definition II.H. requires Type C testing to be performed on a containment isolation valve if it meets at least one of Qualifiers 1, 2, 3 and 4. However, as discussed below, these valves do not meet the qualifiers in Appendix J, Definition II.H., that would require them to be subject to Type C testing.

A large number of small (1" nominal pipe size or less) manual globe valves, including the nine valves which are the subject of this evaluation, are located on test connection, vent, and drain piping which connect with process piping between the inboard and outboard isolation valves of primary containment penetrations. On some penetrations which have both containment isolation valves outside the containment wall, the test connection, vent, or drain piping connects with the process piping between the containment wall and the first (inboard) isolation valve. On some penetrations, the penetration itself is the test connection, and the valves are installed accordingly. All of these valves meet the design requirements of GDC 55 or 56, as noted in UFSAR Table 6.2-44, for designation as containment isolation valves.

These test connection, vent, and drain valves are maintained closed at all times except when they are being used for filling, venting, draining or performing testing which require them to be open. The positions of these valves are controlled and they are verified to be in their closed positions during valve lineup verifications after major system evolutions and prior to restarting the plant. On most of these penetrations, including all of the penetrations that are the subject of this safety evaluation, there is a second isolation valve in series which is also controlled. There is also generally a pipe cap at the end of a test connection or vent line. In addition, as noted in UFSAR Table 6.2-44, all of the test connection, vent, and drain valves listed in UFSAR Tables 6.2-44 and 6.2-49 are required to be locked in the closed position during normal operation.

Applicability of Qualifier 1: Since they are controlled and locked closed during normal operation, these valves do not provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation. Therefore, they do not meet Qualifier 1 of Appendix J, Definition II.H., for Type C testing.

Applicability of Qualifier 2: Since they are manually-operated valves with no provision for power operation, these valves cannot close automatically upon receipt of a containment isolation signal. Therefore, they do not meet Qualifier 2 for Type C testing.

Applicability of Qualifier 3: Many of these valves are located in the drywell or other parts of the plant which would not be accessible under post accident conditions. (The nine valves that are the subject of this safety evaluation are in the Auxiliary Building Steam Tunnel.) All of these valves are on systems which either are required to operate under post-accident conditions or are required to isolate from the containment on receipt of a containment isolation signal. In either case these valves must remain closed and would not be opened during post-accident conditions. Therefore, they do not meet Qualifier 3 for Type C testing.

Applicability of Qualifier 4: These valves are all on systems which penetrate containment of Grand Gulf 1, which is a direct-cycle boiling water power reactor; therefore, on casual inspection, they all meet Qualifier 4 for Type C testing requirements. However, we believe that Qualifier 4 requires interpretation to determine its intent.

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued):**

BASIS (CONTINUED): Qualifier 4 seemingly requires that all containment isolation valves on any system that penetrates containment of a direct-cycle boiling water power reactor (BWR) must be Type C tested. In addition, Qualifier 4 appears to exclude from Type C testing all containment isolation valves on indirect-cycle pressurized water power reactors (PWRs), including valves and systems with similar purposes and designs, unless they also meet Qualifier 1, 2 or 3 of definition II.H. This interpretation does not appear to be the intent of Definition II.H.

The examples given in Definition II.H., "main steam and feedwater piping and other systems which penetrate containment ...," suggest the true intent of Qualifier 4. For the purposes of containment isolation and leakage measurement, a direct-cycle boiling water reactor (BWR) plant differs from an indirect-cycle pressurized water reactor (PWR) plant in one fundamental respect. The BWR main steam, feedwater, and some other steam system pipe lines penetrating the containment boundary carry reactor coolant, whereas the similar-purpose PWR lines carry uncontaminated secondary-side water. The PWR steam generators (SGs) provide a major additional barrier to the escape of highly-contaminated reactor coolant which is not present in a BWR plant. Most of the other BWR systems, such as reactor water cleanup, residual heat removal, and high pressure core spray, have counterparts in PWR systems, such as purification, decay heat removal and high pressure injection, which perform similar functions. In addition, such systems as service air, drain, sampling, and ventilation systems perform the same functions in both BWR and PWR plants.

It is unreasonable to assume that BWR plant systems that are designed and constructed similarly to their PWR plant counterparts and perform essentially-identical functions should be subject to significantly more restrictive leakage testing requirements than their counterparts at PWR plants, unless there is a significant reason.

For the main steam and feedwater systems, the reason is the PWR steam generator, which prevents the turbine steam and feedwater from being contaminated from contact with the reactor core. The PWR main steam and feedwater lines, as well as some auxiliary lines, such as steam drains and steam generator chemical addition lines, carry water and steam which are not radioactively contaminated; therefore, there is not the concern with escape of contaminated fluid from containment that there is with direct-cycle BWR main steam, feedwater, and some other systems. Plant-to-plant differences in the cleanliness and level of contamination in the main steam and feedwater among PWR plants and among BWR plants are not significant compared with the much greater potential for release of radioactivity to the environment through the BWR steam and feedwater lines. It is clear, then, that the intent of Qualifier 4 is to require leakage testing of valves and penetrations at BWR plants which pose a significantly greater hazard to the environment than valves and penetrations which perform similar functions at PWR plants.

Table 1 attached to and part of the justification for UFSAR Change Request No. PL-93-007 (incorporated into this evaluation by reference) compares the penetrations at Grand Gulf Unit 1 with similar-purpose penetrations at PWR plants and identifies those penetrations which meet Qualifier 4.

From a study of Table 1, it is clear that the only penetrations which meet Qualifier 4 are those that carry reactor steam out of containment or that carry feedwater from non-safety related components outside containment to the reactor. The only penetrations at Grand Gulf Unit 1 which carry reactor steam or feedwater into or out of containment are the following:

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued):**

BASIS (CONTINUED):

Penetrations 5-8:	Main Steam Lines A-D
Penetrations 9 & 10:	Feedwater Lines A & B
Penetration 17:	Steam Supply to RCIC Turbine and RHR Heat Exchangers
Penetration 19:	Main Steam Line Drains

It has already been demonstrated that manual test connection, vent and drain valves do not meet Qualifiers 1, 2, or 3 of Appendix J, Definition II.H. Manual test connection, vent and drain valves on penetrations other than the eight penetrations identified above also do not meet Qualifier 4. In accordance with Definition II.H, they do not require Type C testing as containment isolation valves. Therefore, they have been excluded from the Type C leakage testing requirements of GGNS Improved Tech. Spec. SR 3.6.1.1.1 (previously Tech Spec. 4.6.1.2.d through i), as documented in and justified by UFSAR Change Request No. PL-93-007. The change request added notes to UFSAR Tables 6.2-44 and 6.2-49 that clarify that these valves do not require Type C testing. The valves were subsequently removed from the GGNS Type C leakage testing program.

In addition to meeting Qualifier 4 of Definition II.H., the main isolation valves on the eight steam and feedwater penetrations identified above also meet at least one of Qualifiers 1, 2, or 3. Seven of these penetrations (the exception is Penetration 19) also have test connection valves within the containment penetration boundary, which are listed in UFSAR Tables 6.2.44 and 6.2-49 as containment isolation valves. In addition, two feedwater penetrations (Penetrations 9 and 10) also have drain valves within the containment penetration boundary. (Penetration 19 does not have any test connection, vent or drain valves within the penetration boundary.) The test connection valves (a total of nine valves) are currently in the GGNS Type C leakage testing program. These nine valves, which are listed in Section I.B, "Executive Summary," are the subject of this Safety Evaluation.

We believe that the intent of Definition II.H. Qualifier 4 is to require Type C testing of the main isolation valves in applicable piping penetrating primary reactor containment but not to require Type C testing of the test connection, vent and drain valves that may be connected to the piping. This belief is based on the following documents produced and issued by the NRC, and it may reflect a possible shift in thinking by the Nuclear Regulatory Commission (NRC) from their position when Appendix J was initially issued.

1. In 1992, the NRC staff sent a proposed revision to Appendix J to the NRC Commissioners for approval and subsequent publication [See NUMARC (T. E. Tipton) letter, dated January 17, 1992, Subject: "NRC Proposed Revision to 10 CFR 50, Appendix J, Containment Leak Rate Testing]. The 1992 proposed revision to Appendix J contains the following definition of Type C test:

   "Type C Test means a pneumatic test to measure containment isolation valve leakage rates."

   In addition, the NRC's 1992 proposed revision to Appendix J contains the following Section III.C.5.(a):

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued):**

BASIS (CONTINUED):

"5. Valves That Need Not Be Type C Tested.

"(a) A containment isolation valve need not be Type C tested if the valve does not constitute a potential containment atmosphere leakage path during or following an accident, considering the most limiting single active failure."

The Bases above in this evaluation have adequately established that, based on qualification of their design, materials and construction, and on multiple barries to leakage maintained closed during power operation under an administrative control program, these test connection valves do not constitute a potential containment atmosphere leakage path during or following an accident.

2. The NRC's 1992 proposed revision to Appendix J refers to a draft Regulatory Guide 1.XXX (number to be assigned) for "Specific guidance concerning acceptable leakage rate test methods, procedures, and analyses ... ." The draft regulatory guide did not accompany the 1992 proposed Appendix J revision; however, the accompanying discussion material indicated that the regulatory guide would endorse the 1987 revision of ANS 56.8 and would be revised to endorse any later revisions of ANS 56.8.

The 1987 approved revision to ANSI/ANS 56.8, American National Standard, "Containment System Leakage Testing Requirements," addresses test connection valves. In Section 6.2, "Test Boundaries and Connections for Testing," the standard states the following:

"If it is necessary to install test connections between redundant containment isolation valves, the connection should consist of a double barrier (e.g., two valves in series, one valve with a nipple and cap, or one valve with a nipple and blind flange). These test connections are part of the containment system barrier, but due to their infrequent use and multiple barriers, they do not require leakage rate testing as long as the barrier configurations are maintained using an administrative control program."

All of the nine test connection valves that are the subject of this evaluation have a triple barrier of two valves in series with a nipple and pipe cap. In addition, all of these test connections join to their process pipes in the Auxiliary Building Steam Tunnel outboard from the inboard main containment isolation valves. Therefore, all of these nine test connection valves meet or exceed the requirements of ANSI/ANS 56.8-1987 for exemption from Type C testing.

3. The NRC staff prepared a Draft Regulatory Guide, Task DG-1037 in August, 1994, entitled "Performance-Based Containment Leak Test Program." In Section C, "Regulatory Position," the NRC staff proposed to endorse Draft Nuclear Energy Institute (NEI) Guideline NEI 94-01, Revision D, dated October 25, 1994, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J." NEI Guideline 94-01 contains the following words under Section 6.0 General Requirements on Page 4:

"An LLRT is a test performed on Type B and Type C components. An LLRT is not required for the following cases:

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued):**

BASIS (CONTINUED):

"• Primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA);

"• Boundaries sealed with a qualified seal system; or,

"• Test connection vents and drains between primary containment isolation valves which are one inch or less in size, administratively secured closed and consist of a double barrier."

All of the nine test connection valves that are the subject of this evaluation are 3/4-inch in nominal pipe diameter and have a triple barrier of two valves in series with a nipple and pipe cap; therefore, they meet the third bulleted qualifier above. In addition, since they are administratively controlled closed at all times during power operation, they also meet the first bulleted qualifier above.

The regulatory position in the Draft Regulatory Guide DG-1037 took exception to certain guidelines in NEI 94-01, but it did not take exception to the guidance quoted above, thereby implying NRC acceptance of the guidance.

The points above clearly establish that, since at least 1992, the NRC staff has been willing to accept the position that Appendix J does not require these nine test connection valves to be Type C tested. Based on the points above, Appendix J, as currently interpreted by the NRC, does not require these nine test connection valves to be Type C tested.

The accident scenarios in the UFSAR assume no leakage through test connection, vent and drain valves; however, administrative controls on valve position provide reasonable assurance of no leakage through any test connection, vent or drain lines without the need to perform leakage testing. These administrative controls ensure that at least a double barrier to leakage is present at all times when containment integrity is required. The following are the administrative controls which are considered in this evaluation:

a. Explicit restoration instructions in the local leak rate test procedures for containment penetrations specify the restored positions for the valves and pipe caps on each test, vent or drain connection used in local leak rate tests and require double verification that the valves and caps are correctly restored. All of these nine test connections have three barriers (two manual globe valves and a pipe cap) in series.

b. These penetrations and valves are lined up and verified to be in specified positions when restoring each system to operability, in accordance with system operating instructions.

c. These test connection valves, since they are located in the Auxiliary Building Steam Tunnel, are verified closed during each cold shutdown, if not verified within the previous 92 days. The verification is normally performed immediately before, or in conjunction with, startup of the plant from the cold shutdown condition.

d. Pipe caps on these penetrations are controlled under the GGNS configuration control program.

2. May increase the consequences of an accident previously evaluated in the SAR **(Continued)**:

BASIS (CONTINUED): Therefore, the consequences of an accident previously evaluated in the UFSAR are not increased by these changes.

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR. ☐ Yes ☒ No

BASIS: The Appendix J leakage testing requirements are intended only to minimize leakage of radioactivity from containment during and following an accident.

The occurrence and consequences of malfunctions of equipment important to safety could be affected by leakage through these valves; however, leakage rate testing of these valves could not be relied on to detect such leakage, since the valves are usually repositioned after the leakage rate tests. Improper valve positioning is the most likely cause of any such leakage.

The probability of improper valve positioning is minimized by performance of valve alignments and position verifications under the Administrative Controls program described in Section II.B.2 above. Leak rate testing of these valves is not designed to identify mispositioned valves. Therefore, administrative control of valve position, not leak rate testing, is the best way to limit leakage due to mispositioning of these valves.

Damage to the valve's seat or disk is the only physical damage that could be detected only by leak rate testing these test connection valves. The leak rate tests of the main isolation valves in each penetration through these test connection valves check for leakage through the pipe on both sides of the test connection valve, as well as through the valve body and stem packing. Damage to the seat or disk of one of these test connection valves is unlikely for the following reasons:

a. All of the penetrations, pipe and valves are designed and constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB or NC (Class 1 or 2).

b. These test connection valves are used only for leak rate testing other valves in their respective penetrations. They are not subjected to flow during plant operation, and are normally closed, except during tests. During leak rate tests of the other valves, the only flow through the valves is clean, filtered air or water.

c. These test connection valves may also be used for venting, draining and filling their respective process piping during cold shutdown conditions. They are not subjected to flow during plant operation, and are normally closed, except during venting, draining and filling. Under these conditions, the systems are not at high pressure or temperature; therefore, the danger of significant damage to the seats and disks of these valves during venting, draining and filling is minimal. In addition, such venting, draining and filling operations are infrequently performed events (usually no more than once per refueling cycle), and, therefore, do not subject the valves to high numbers of cycles.

Therefore, this change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: The Appendix J leakage testing requirements are intended only to minimize leakage of radioactivity from the containment during and following an accident. The occurrence and consequences of malfunctions of equipment important to safety could be affected by leakage through these valves; however, leakage rate testing of these valves could not be relied on to detect such leakage, since the valves are usually repositioned after the leakage rate tests.

Improper valve positioning is the most likely cause of any such leakage. The probability of improper valve positioning is minimized by performance of valve alignments and position verifications under the Administrative Controls program described in Section II.B.2 above. Leak rate testing of these valves is not designed to identify mispositioned valves. Therefore, administrative control of valve position, not leak rate testing, is the best way to limit leakage due to mispositioning of the test connection valves.

Damage to the valve's seat or disk is the only physical damage that could be detected only by leak rate testing these test connection valves. The leak rate tests of the main isolation valves in each penetration through these test connection valves check for leakage through the pipe on both sides of the test connection valve, as well as through the valve body and stem packing. Damage to the seat or disk of one of these test connection valves is unlikely for the following reasons:

a. All of the penetrations, pipe and valves are designed and constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB or NC (Class 1 or 2).

b. These test connection valves are used only for leak rate testing other valves in their respective penetrations. They are not subjected to flow during plant operation, and are normally closed, except during tests. During leak rate tests of the other valves, the only flow through the valves is clean, filtered air or water.

c. These test connection valves may also be used for venting, draining and filling their respective process piping during cold shutdown conditions. They are not subjected to flow during plant operation, and are normally closed, except during venting, draining and filling. Under these conditions, the systems are not at high pressure or temperature; therefore, the danger of significant damage to the seats and disks of these valves during venting, draining and filling is minimal. In addition, such venting, draining and filling operations are infrequently performed events (usually no more than once per refueling cycle), and, therefore, do not subject the valves to high numbers of cycles.

Therefore, this change does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: The Appendix J leakage testing requirements are intended only to minimize leakage of radioactivity from the containment during and following an accident. They do not create the possibility of a different type of accident.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: Any accident that could be possible due to leakage through one or more of the valves considered in this change would be due to improper positioning of the valve(s). Leakage testing of these valves is not designed to identify mispositioned valves. The probability of improper valve positioning is minimized by control of and performance of valve alignments and position verifications under the Administrative Controls program described in Section II.B.2 above.

The Appendix J leakage testing requirements are intended only to minimize leakage of radioactivity from the containment during and following an accident. Various malfunctions of the containment isolation system components have already been analyzed for such accidents.

These test connection valves are not being physically modified or deleted. These changes delete leak rate testing of these test connection valves. Leakage through these valves has already been considered in loss of containment integrity accidents that have already been analyzed in the UFSAR. Therefore, these changes do not create the possibility of a malfunction of a different type than any evaluated previously in the UFSAR.

7. Will reduce the margin of safety as defined in the basis for any Technical Specification.

☐ Yes
☒ No

BASIS: Deleting the Type C leak rate test requirements for these valves does not affect the operation or operating parameters of any system or the plant. Leak rate testing is only a precaution. These changes do not physically add, modify or delete any component or setpoint in the plant. These changes delete leak rate testing of these test connection valves on main steam and feedwater penetrations. They do not add, modify or remove from the plant any component required to be leak rate tested.

The 1987 approved revision to ANSI/ANS-56.8, American National Standard "Containment System Leakage Testing Requirements," addresses these test connection valves. In section 6.2, "Test Boundaries and Connections for Testing," the standard states the following:

> "If it is necessary to install test connections between redundant containment isolation valves, the connection should consist of a double barrier (e.g., two valves in series, one valve with a nipple and cap, or one valve with a nipple and blind flange). These test connections are part of the containment system barrier, but due to their infrequent use and multiple barriers, they do not require leakage rate testing as long as the barrier configurations are maintained using an administrative control program."

All of the test connection valves identified in these changes have a triple barrier of two valves in series and a pipe cap. In addition, these test connections join to their process pipes between inboard and outboard isolation valves. This main process valve provides an additional barrier to leakage through the penetration.

Additionally, ASME Boiler and Pressure Vessel Code, Section XI, Subarticle IWV-1200, exembts these valves from inservice testing, including the seat leakage testing which would otherwise be required by Subarticle IWV-3420, due to their being one inch nominal pipe size or smaller.

7. Will reduce the margin of safety as defined in the basis for any Technical Specification. **(Continued):**

BASIS (CONTINUED): The margins of safety defined in Improved Tech Spec LCO 3.6.1.1 and SR3.6.1.1.1 (previously Tech Specs 3/4.6.1.1 and 3/4.6.1.2) Bases could only be reduced if significant leakage would not be detected except by leak rate testing. Such leakage could only be due to mispositioned valves or physical damage to the pressure boundary. The probability of improper valve positioning is minimized by performance of valve alignments and position verifications under the Administrative Controls program described in Section II.B.2 above. Leak rate testing of these valves is not designed to identify mispositioned valves.

In addition, these valves may be opened to fill and vent piping during system restoration after they are tested. Therefore, administrative control of valve position, not leak rate testing, is the best way to limit leakage due to mispositioning test connection valves.

Damage to the valve's seat or disk is the only physical damage that could be detected only by leak rate testing these test connection valves. The leak rate tests of the main isolation valves in each penetration through these test connection valves check for leakage through the pipe on both sides of the test connection valve, as well as through the valve body and stem packing. Damage to the seat or disk of one of these test connection valves is unlikely for the following reasons:

a. All of the penetrations, pipe and valves are designed and constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB or NC (Class 1 or 2).

b. These test connection valves are used only for leak rate testing other valves in their respective penetrations. They are not subjected to flow during plant operation, and are normally closed, except during tests. During leak rate tests of the other valves, the only flow through the valves is clean, filtered air or water.

c. These test connection valves may also be used for venting, draining and filling their respective process piping during cold shutdown conditions. They are not subjected to flow during plant operation, and are normally closed, except during venting, draining and filling. Under these conditions, the systems are not at high pressure or temperature; therefore, the danger of significant damage to the seats and disks of these valves during venting, draining and filling is minimal. In addition, such venting, draining and filling operations are infrequently performed events (usually no more than once per refueling cycle), and, therefore, do not subject the valves to high numbers of cycles.

Therefore, these changes do not reduce the margins of safety defined in the bases for Improved Tech Spec LCO 3.6.1.1 and SR3.6.1.1.1 (previously Tech Specs 3/4.6.1.1 and 3/4.6.1.2).

III. Environmental Evaluation

☐ Not applicable per Environmental Evaluation Applicability Review

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

A. Environmental Protection Plan

1. Will require a change in the Environmental Protection Plan. ☐ Yes ☒ No

BASIS: The Type C testing program is not addressed in the Environmental Protection Plan.

B. Unreviewed Environmental Question

1. Concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB. ☐ Yes ☒ No

BASIS: As long as the primary containment is maintained leak-tight within the requirements of 10 CFR 50, Appendix J, potential radioactive releases to the environment as a result of accidents will not exceed the guidelines of 10 CFR 100, and there will be no negative effect on the environment.

The proposed changes to the Type C testing program have been evaluated as discussed in Sections I (Change to Technical Specifications) and II.A (Unreviewed Safety Question) above and have been determined to be acceptable, as discussed in the Basis statements above.

These changes do not change any physical plant system or component, nor do they change any operating or accident response procedures described in the UFSAR or Technical Specifications.

Since the overall containment leakage rates will be maintained within the limits allowed by 10 CFR 50, Appendix J, the GGNS Technical Specifications, and the GGNS UFSAR, even with these changes, there will be no effect on the environment.

2. Concerns a significant change in effluents or power level. ☐ Yes ☒ No

BASIS: The test connection valves addressed in this evaluation are closed at all times during power operation, and are opened only for testing, venting, draining, and filling the associated piping systems. All effluents that may be collected from these drains are handled in accordance with plant administrative procedures and the Health Physics program.

These changes do not represent any physical changes in the plant or in any plant operating procedure that could affect effluents or power level. Discussion under Sections II.B.1 through II.B.4 above clearly explain why the probability of occurrence and consequences of accidents and equipment malfunctions, which could increase effluents, will not be increased.

3. Concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1 above, which may have a significant environmental impact. ☐ Yes ☒ No

BASIS: As long as the primary containment isolation valves are maintained within the maximum allowable leakage rates addressed in the UFSAR, potential radioactive releases to the environment as a result of accidents will not exceed the guidelines of 10 CFR 100, and there will be no negative effect on the environment.

The proposed changes to the Type C testing program have been evaluated as discussed in Sections I (Change to Technical Specifications) and II.A (Unreviewed Safety Question) above and have been determined to be acceptable, as discussed in the Basis statements above.

These changes do not change any physical plant system or component, nor do they change any operating or accident response procedures described in the UFSAR or Technical Specifications.

Since the overall containment leakage rates will remain within the limits required by 10 CFR 50, Appendix J, there will be no effect on the environment.

Signatures and Approvals

Evaluated: A. J. Malone 8-28-95
ORIGINATOR / DATE

Reviewed/Approved: [signature] 8/28/95
REVIEWER / DATE

Plant Safety Review Committee Review

[signature] 9/2/95
CHAIRMAN, PSRC / DATE

95-098-PSE

GRAND GULF NUCLEAR STATION UNIT 1
CHANGES, TESTS OR EXPERIMENTS SAFETY AND ENVIRONMENTAL EVALUATION FORM

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: Ken Walker DEPT/SECT: P&SE/RE EVAL. #: 95-0072-R00

DOCUMENT EVALUATED: EER 95-6156

REFERENCES: EER 95-6156; GTC - 95/0299; GEK 73674A

FSAR CHANGE REQUIRED? ☐ Yes ☒ No CR # NA

FSAR SECTIONS TO BE REVISED: NA

TRM CHANGE REQUIRED? ☐ Yes ☒ No

TECH. SPEC. CHANGE REQUIRED? ☐ Yes ☒ No CR # NA

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? ☒ Yes ☐ No

Validity of the SE is dependent upon Operations procedure changes being in compliance with the stipulations of EER 95-6156. Specifically, drive water pressure may not exceed 350 psid above reactor pressure if reactor power is greater than the high power setpoint. Other stipulations are made in the EER, and this SE also relies on those even though they may not be specifically discussed herein.

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? ☒ Yes

THE RESPONSIBLE ORGANIZATIONS MUST BE NOTIFIED PRIOR TO IMPLEMENTING THIS CHANGE.

B. Executive Summary (ALSO SERVES AS INPUT TO NRC SUMMARY REPORT)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: EER 95-6156 allows control rod drive system drive water pressure to be temporarily increased up to 475 psi above reactor pressure for purposes of withdrawing control rods which will not move at normal pressure. This pressure increase is to be done under Operations Off-normal Event Procedures, and is subject to certain limitations as described in the EER and in this evaluation. No FSAR changes are required as the FSAR discusses only normal CRD system pressures and does not discuss limits on temporarily exceeding such pressures.

REASON FOR CHANGE, TEST OR EXPERIMENT: Control rods often prove difficult to withdraw at normal control rod drive system drive water pressure, especially from position 00 and/or during restart from a reactor scram. It is necessary to allow plant Operators to temporarily increase this pressure in order to withdraw the rods. Current procedures (ONEP 05-1-02-IV-1) allow an increase only to 350 psi above reactor pressure. This often is insufficient to initiate movement, and a higher pressure is needed.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: No change to Technical Specifications or the TRM is necessary since the drive water pressure is not specifically discussed in these documents. Discussions with the NSSS supplier, GE, (GTC-95/00299) indicate that drive water pressures up to 500 psi above reactor pressure are allowable without presenting a risk of damage to the CRD mechanism such that control rod scram could be inhibited. One important consideration involves the chance for inadvertent over-notching of the CRD mechanism during withdrawals due to use of elevated pressures. Engineering evaluations by System Engineering and experience indicate that over-notching of up to 4 notches (08 positions) is feasible. If reactor power is above the high power setpoint (~70% power), such an event could result in movement exceeding that intended to be allowed by the Rod Withdrawal Limiter (e.g. 2 notches), thus violating the assumptions of the Rod Withdrawal Error (RWE) analysis. This evaluation therefore supports only the use of elevated pressures above 350 psid when below the high power setpoint. Below that power level, pressure increases up to 475 psid over reactor pressure are temporarily allowable since overnotching of 3 notches, or even 4 notches, would not exceed the RWL/RWE travel limit of 4 notches.

Thus, with the above restrictions, there is no increase in the probability or consequences of any accident or malfunction previously analyzed. The RWL will continue to protect against the possibility of an unbounded Rod Withdrawal Error. The possibility or consequences of a Control Rod Drop Accident are also unaffected by this change. No new types of events are created. There are no additional changes to the system operating procedures nor are there any changes in system design. The control rod drive system will not be inhibited from performing its scram insertion function if called upon to do so. No margins of safety are being affected since there is no impact on compliance with the MCPR safety limit, plastic strain limit, or radiological dose limits. Therefore, no unreviewed safety question is created by allowing an increase in CRD system drive water pressure of up to 475 psid above reactor pressure provided pressure is not increased over 350 psid above reactor pressure when above the high power setpoint.

II. Safety Evaluation

☐ Not applicable per Safety Evaluation Applicability Review

A. Technical Specifications

1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications.

☐ Yes
☒ No

BASIS: The drive water pressure differential is not limited or discussed in the Technical Specifications. The restrictions being placed upon elevated pressure differentials when reactor power is above the high power setpoint will ensure that the basis for TS 3.3.2.1 are maintained. Control rod scram times as required by TS 3.1.4 will not be affected by the proposed change. Any problems with rods which are immovable will continue to be addressed adequately by TS 3.1.3. Any minor rod malfunctions resulting in rod pattern violations will be adequately addressed by TS 3.1.6.

B. Unreviewed Safety Question

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

1. May increase the probability of occurrence of an accident previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: The events in the SAR which are potentially impacted by this change include all events relying upon a reactor scram for mitigation as well as the Rod Withdrawal Error and the Control Rod Drop Accident. Discussions with the NSSS supplier (GTC 95/00299) indicate that drive water pressures of up to 500 psid above reactor pressure are acceptable without damage potential even though these are not specifically allowed by the vendor manual. Relief valves are also provided on the drive water piping to prevent exceeding potentially damaging pressures. Thus, the ability of the control rod drive to accomplish the scram function will not be affected and the potential for a failure to scram when required is not increased.

The Rod Withdrawal Error analysis (UFSAR 15.4.2) considers the inadvertent continuous withdrawal of a control rod to a point where fuel damage could occur. The possibility of unintentional rod withdrawal under limiting conditions is not affected by having a higher drive pressure. (The consequences are discussed under B.2 below.) Further, the Control Rod Drop Accident addresses the results of an uncoupled control rod falling out of the core under worst case conditions. The likelihood of a control rod becoming uncoupled and dropping is not impacted by the proposed change.

2. May increase the consequences of an accident previously evaluated in the SAR. ☐ Yes ☒ No

<u>BASIS</u>: As described above, the ability of the CRD system to perform a reactor scram is not impacted by having a drive water pressure up to 475 psid above reactor pressure since no physical damage can result from such pressures. Should a scram occur while driving a rod, accumulator pressure will still be directed to the underpiston area as designed. Thus, the consequences of any event requiring a reactor scram are not increased.

Concerning the RWE event, two conditions must be considered: <u>above</u> and <u>below</u> the high power setpoint (HPSP). **Above** the HPSP, the RWE analysis (UFSAR 15.4.2) assumes that rod motion is stopped prior to a rod exceeding 1 foot (2 notches) of travel. It is conceivable based on experience and engineering estimates, that inadvertent withdrawals of up to as much as 4 notches could occur at the pressures under consideration. This would violate the assumptions of the RWE analysis even with the withdrawal limiter functioning properly. Thus, this evaluation <u>does not support</u> elevated drive water pressures exceeding the 350 psid value already allowed by ONEP 05-1-02-IV-1 <u>when reactor power is above the HPSP</u>. When power is **below** the HPSP, a 2 foot (4 notch) withdrawal is allowed prior to stopping movement. As discussed in EER 95-6156, experience, judgment, and engineering estimates indicate that over-notching in excess of this amount is not expected at the proposed differential pressure of 475 psid. Further, while rod withdrawal speed is expected to be faster for higher drive water pressure, this is not a factor affecting the RWE analysis results. Thus, should an inadvertent rod withdrawal occur under limiting or near-limiting core conditions, the assumptions of the RWE analyses would be met and the resulting consequences would be no more severe. The MCPR safety limit would not be exceeded, nor would the 1% plastic strain limit be violated. Note that this evaluation assumes that normal procedural controls applicable to operation of the CRD system and RWL which are credited in the RWE analysis continue to apply (UFSAR 15.4.2.2.2).

The consequences of a CRDA are also not increased by higher drive water pressure. The radiological consequences (fuel failures) resulting from the CRDA are influenced primarily by the control rod worth which is a function the core conditions and rod pattern at the time the event occurs. A higher drive pressure does not affect these items except that it is possible for an overnotching event to result in a rod pattern which temporarily violates the constraints of the banked position withdrawal sequence. Such a condition is already anticipated by the Technical Specifications (TS 3.1.6, Action A), however, and is considered within the bounds established by the RWE analysis as discussed in the TS Basis. Postulated system failures in conjunction with a higher drive pressure could result in an unplanned withdrawal at higher than normal (3 in/sec) speeds, however such speeds still remain less than those assumed in the CRDA (UFSAR 4.6.2.3.2.2) and are therefore bounded by the existing analysis.

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: No other modifications to equipment or procedures important to safety are being made by this change. Discussions with GE personnel indicate that the system is capable of handling the increased pressure without adverse affects. A higher drive pressure will result in a slightly increased speed of withdrawal or insertion, however this is not considered a "malfunction" since it is expected and can occur even with normal variations in system pressure. Examination of the system design also shows that increased drive water pressure is not more likely to result in a unplanned withdrawal or insertion without the presence of additional failures.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: The FSAR considers several malfunctions of the CRD system (UFSAR 4.6.2.3). The proposed change potentially impacts only the consequences of an unplanned withdrawal assuming additional failures occur. The consequences of other events are either unrelated or negligibly affected by drive pressure prior to an assumed failure. In the case of an unplanned withdrawal, a slightly faster withdrawal speed could result, however this remains bounded by existing CRDA analyses.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: Increasing drive water pressure to a maximum of 475 psid above reactor pressure does not create the possibility of an accident of a different type. No physical changes are being made to system design, and no additional changes are being made to system operation. The UFSAR already considers numerous malfunctions, ruptures, and multiple failures of the CRD system and its components. This relatively slight increase in allowed drive water pressure under controlled conditions is well within system piping design capacity. Any impact on rod mispositioning is already considered as described in B.1 and B.2 above.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

☐ Yes
☒ No

BASIS: As described in B.5 above, the UFSAR already considers a large spectrum of events related to CRD system malfunctions. No additional malfunctions which could be created by an increased drive flow could be identified.

7. Will reduce the margin of safety as defined in the basis for any Technical Specification.

☐ Yes
☒ No

BASIS: No margin of safety is impacted. With the restriction placed on excessive pressures above the HPSP, there is no danger of exceeding the MCPR safety limit or the 1% plastic strain limit discussed in the basis for TS 3.3.2.1. The 280 cal/gm limit related to the CRDA event discussed in TS 3.1.6 basis is likewise not threatened by the proposed change since no change is being made to existing rod pattern controls or associated Technical Specifications. Also, control rod scram ability and times will not be affected by the proposed change so assumed reactivity insertion margins remain the same.

There is no evidence that the NRC made any assumptions regarding drive water differential pressure in the conclusions described in the GGNS SER. One mention of an NRC concern regarding failure effects on drive water pressure and resulting speed (SER, Sect. 4.0) is resolved in the UFSAR analysis.

III. Environmental Evaluation

☐ Not applicable per Environmental Evaluation Applicability Review

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

A. Environmental Protection Plan

1. Will require a change in the Environmental Protection Plan.

☐ Yes
☒ No

BASIS: The CRD system is not discussed in the EPP.

B. Unreviewed Environmental Question

1. Concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB.

☐ Yes
☒ No

BASIS: The proposed change has no potential impact on the environment in excess of that already considered in the FES.

2. Concerns a significant change in effluents or power level..

☐ Yes
☒ No

BASIS: No change to effluents or power level.

3. Concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1 above, which may have a significant environmental impact.

☐ Yes
☒ No

BASIS: The proposed change does not present a possible significant environmental impact.

Signatures and Approvals

Evaluated: Ken L. Walker 8/1/95
ORIGINATOR / DATE

Reviewed/Approved: Lee E. Eaton 8-1-95
REVIEWER / DATE

Plant Safety Review Committee Review

8-1-95
CHAIRMAN, PSRC / DATE

95-101-Chem

GRAND GULF NUCLEAR STATION UNIT 1

CHANGES, TESTS OR EXPERIMENTS SAFETY AND ENVIRONMENTAL EVALUATION FORM

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: J.M.LASSETTER DEPT SECT: CHEM EVAL. #: 95-0078-R01

DOCUMENT EVALUATED: TRM c/r 95-071

REFERENCES: TRM LCO 6.3.10, 6.3.11, NUREG 0578, 0737, REG GUIDE 1.97, UFSAR CHAPTER 11, 15, 18, LICENSING RESEARCH SYSTEM (LRS), FINAL ENVIRONMENTAL STATEMENT.

FSAR CHANGE REQUIRED? ✓ Yes ☐ No CR # 95-072

FSAR SECTIONS TO BE REVISED: 11.5.2.2.4.1(editorial change)

TRM CHANGE REQUIRED? ✓ Yes ☐ No

TECH. SPEC. CHANGE REQUIRED? ☐ Yes ✓ No CR # CR or (n/a)

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? ✓ Yes ☐ No

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? ✓ Yes

THE RESPONSIBLE ORGANIZATIONS MUST BE NOTIFIED PRIOR TO IMPLEMENTING THIS CHANGE.

B. Executive Summary (ALSO SERVES AS INPUT TO NRC SUMMARY REPORT)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: TRM LCO 6.3.11 Action B.1 is being deleted. This requires re-numbering action, "B.2", to "B.1" and changing the referenced Action in Condition "C" from "B.2" to "B.1". Existing Action B.1 requires, within 72 hours, implementation of pre-planned alternate method of monitoring the appropriate parameters when the accident range (AXM) noble gas radiation monitor function is inoperable.

This action may be deleted because it is bounded by a more limiting TRM LCO. TRM LCO 6.3.10 requires continuous monitoring of gaseous radioactive effluents, contains compensatory measures for inoperable equipment and applies to the same pathways and parameters (functions) described in the action to be deleted (TRM LCO Action 6.3.11, Action B.1). TRM LCO 6.3.10 requires compensatory actions (grab sampling and analysis) in a shorter time frame (four or eight hour sample collection, analysis within 24 hours) than TRM LCO 6.3.11(72 hours).

In summary, the proposed activity is deletion of a redundant requirement. TRM LCO 6.3.10 Actions B and G. require responses which are adequate for preplanned, alternate monitoring methods. An editorial UFSAR change will be submitted to specify that provisions exist for collection of grab samples in the event of a loss of Accident Range Monitoring instrumentation (UFSAR c/r 95-072).

REASON FOR CHANGE, TEST OR EXPERIMENT: A redundant TRM LCO action exists (6.3.11.B.1). Other TRM requirements (LCO 6.3.10 Action B.1,B.2 and G.1, G.2 provide more restrictive compensatory measures which meet the requirements of the deleted action.

The requirement to continually monitor gaseous radioactive effluents is independent of accident or non-accident conditions.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: The proposed change will not result in a change to the Radiological Effluent Controls Program as required by Technical Specification 5.5.4.

The associated D17 Noble Gas Effluent monitors (Eberline SPING for SBGT-A & B, General Electric for OffGas/RadWaste, Containment, Turbine and Fuel Handling) are always the pre-planned alternate method of monitoring provided they are operable and onscale. If these monitors are inoperable, the pre-planned alternatives are procedurally in place as required by TRM LCO 6.3.10. The pre-planned alternative monitoring methods are in fact always in effect and passive in nature. No additional actions are required.

Regulations requiring accident range noble gas monitoring instrumentation are based on inadequacies identified following the TMI-2 accident. The regulations reviewed for this safety evaluation contain specifications for installed noble gas effluent monitors but do not delineate actions required when those instruments are inoperable. Obviously, periods of inoperability are expected to be minimized. The TRM requirement to return noble gas accident range monitoring channel(s) to OPERABLE status within seven days and associated reporting provisions are unaffected by the proposed change.

II. Safety Evaluation ☐ Not applicable per Safety Evaluation Applicability Review

A. Technical Specifications

1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications. ☐ Yes ✓ No

   BASIS: The affected requirements are not in the Technical Specifications. No change to the Radiological Effluent Controls Program as described in the TS will result from the proposed change. The affected requirement : alternate, pre-planned methods of monitoring of radioactive noble gas effluents when accident range Instrumentation is inoperable, will utilize existing procedures which fulfill the commitment for routine monitoring of radioactive noble gas effluents.

B. Unreviewed Safety Question

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

1. May increase the probability of occurrence of an accident previously evaluated in the SAR.

☐ Yes
✓ No

BASIS: The proposed activity has no effect on the probability of accidents as described in Chapter 15 of the UFSAR. Accident probability is determined in part by the operating characteristics of installed equipment and the procedures and programs used to maintain the equipment. Installed equipment is unaffected by the proposed activity as the proposed activity is limited to compensatory measures for inoperable gaseous effluent monitoring equipment.

2. May increase the consequences of an accident previously evaluated in the SAR.

☐ Yes
✓ No

BASIS: Radiological consequences from previously evaluated accidents are not affected by the proposed activity. The compensatory actions which result from inoperable equipment continue to ensure adequate monitoring capability of noble gas effluents for long term assessment of Offsite releases. Radiological source terms are not affected nor are existing margins of safety for any plant structures, systems or components.

GGNS UFSAR Section 11.5.2.2.4.1 states that in the case of inoperable gaseous effluent monitoring equipment, provisions have been made to obtain grab samples for laboratory analysis. The proposed change does not affect the actions which fulfill this commitment.

Section surveillance procedures for inoperable effluent gaseous radiation monitors will include specific actions and precautions regarding grab sample collection of noble gas samples under accident conditions.

3. May increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

☐ Yes
✓ No

BASIS: The proposed change is restricted to compensatory measures for inoperable equipment . These measures are independent of equipment maintenance , calibration, and surveillance activities , therefore the probability of that equipment malfunctioning cannot increase as a result of the proposed change. Accident Range monitoring is based on NUREG 0737 requirements. USFAR Section 18.1.27.1 contains GGNS' response to the Noble Gas Monitoring requirements of NUREG 0737. The proposed change does not affect GGNS' response.

4. May increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.
☐ Yes
✓ No

BASIS: The radiological consequences of equipment malfunction will not be affected by alternate monitoring methods provided for inoperable equipment. The proposed change is independent of equipment operating, maintenance and calibration procedures. The requirement to restore inoperable channel(s) to OPERABLE status with seven days (TRM LCO 6.3.11, Action B.2) and subsequent reporting requirements are unaffected by the proposed change.

5. May increase the possibility for an accident of a different type than any previously evaluated in the SAR.
☐ Yes
✓ No

BASIS: The proposed activity is independent of accident types. The proposed change will not result in a change to plant configuration or operating procedures . The proposed change is restricted to methods for monitoring noble gas releases when accident range monitors are inoperable.

6. May create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.
☐ Yes
✓ No

BASIS: The proposed change will not affect plant Structures Systems or Components. No additional functions are required as a result of the proposed change. Equipment will not be operated in a manner other than previously described in the UFSAR.

7. Will reduce the margin of safety as defined in the basis for any Technical Specification.
☐ Yes
✓ No

BASIS: The proposed activity will not result in changes to operating parameters associated with plant structures, systems or components. The proposed change is restricted to actions required to compensate for inoperable accident range equipment and will utilize existing methodology. Requirements to return channel(s) to OPERABLE status are unaffected by the proposed change.

III. Environmental Evaluation

☐ Not applicable per Environmental Evaluation Applicability Review

IMPLEMENTATION OR PERFORMANCE OF THE ACTION DESCRIBED IN THE EVALUATED DOCUMENT:

A. Environmental Protection Plan

1. Will require a change in the Environmental Protection Plan. ☐ Yes ✓ No

BASIS: The proposed activity is restricted to existing equipment and procedures and will not affect the EPP. The EPP deals with non-radiological issues, the proposed activity is limited to monitoring of radiological effluents.

B. Unreviewed Environmental Question

1. Concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB. ☐ Yes ✓ No

BASIS: The proposed activity is a clarification of methods used to compensate for inoperable accident range effluent noble gas monitors. Existing TRM LCO actions for routine effluent monitoring require periodic collection of grab samples upon loss of continuous monitoring capability. These same actions, with consideration for accident range release rates are adequate for inoperable accident range gaseous effluent monitors. The FES does not contain specific requirements for compensatory monitoring methods. FES section 4.2.5 does state that routine measurement of all principal release points is required. The proposed change does not contradict this FES statement.

2. Concerns a significant change in effluents or power level.. ☐ Yes ✓ No

BASIS: The proposed activity will not increase emissions from GGNS. The proposed change is limited to monitoring of radioactive effluents and does not affect parameters associated with generation of those effluents. No reduction in measurement accuracy or increase in sampling periodicity will result from the proposed activity. The nature of the proposed activity excludes any effect on power level.

3. Concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1 above, which may have a significant environmental impact. ☐ Yes ✓ No

BASIS: The proposed activity is restricted to previously disturbed areas . The proposed activity is limited to exiting procedures and equipment. No environmental impact will result from the proposed activity.

Signatures and Approvals

Evaluated: J M Canette 9.19.95

ORIGINATOR / DATE

Reviewed/Approved: Jane Antoine 9-20-95

REVIEWER / DATE

Plant Safety Review Committee Review

N P Farris 9/28/95

CHAIRMAN, PSRC / DATE